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August 13, 2004

U.S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
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Subject: USNRC Docket No. 72-1014, TAC L23657  
HI-STORM 100 Certificate of Compliance 1014  
HI-STORM License Amendment Request 1014-2, Revision 2, Supplement 5

References: 1. Holtec Project 5014  
2. Meeting between USNRC and HOLTEC on August 4, 2004

Dear Sir:

As committed in the meeting on August 4<sup>th</sup>, we herewith provide Supplement 5 to License Amendment Request (LAR) 1014-2, Revision 2. In this supplement, the allowable MPC decay heat loads were reversed to the levels approved in Revision 1 of the CoC, while retaining all other changes previously submitted under this LAR. The most important of these retained changes are

- Changes to support compliance with ISG 11
- Changes to support compliance with ISG 18
- Additional Neutron Absorber Material
- Clarification of ZPA in Cask Sliding
- Various improvements of the approved content
- Changes in soluble boron requirements
- Changes to address potential presence of ice or water under the HI-STORM

The following attachments are provided.

Attachment 1: Summary of Proposed Changes: This document has been appropriately revised to reflect the modifications. All changes to this document from the previously submitted summary are clearly marked.

Attachment 2: Mark-Up of Proposed CoC changes.

Attachment 3: Final Version of Proposed CoC changes.

Attachment 4: Revised FSAR. This is a full copy of the FSAR including all revised and unchanged sections and figures. If there was any change to a previously submitted section, either individual pages or the entire section is provided with "Proposed Revision 2E" in the footer. Note that sections or figures included from Revision 1 still might show revision bars. These are related to the changes in Revision 1, not to changes in this LAR. The current revision level of each FSAR page and figure is provided in an



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updated List of Effective Pages. An updated Table of Contents is also provided. All changes proposed against FSAR Revision 1 are identified in italic/strikeout format.

We appreciate the SFPO's continued diligent review of this amendment. Certain elements of this amendment request are required by our users for deployment of dry storage systems in the near future. To that end, we request approval of this amendment (rulemaking complete) by December 31, 2004. If you have any questions or require additional information, please contact us.

Sincerely,

Evan Rosenbaum, P.E.  
Project Manager, LAR 1014-2

Approval:

Stefan Anton, Dr.-Ing.  
Acting Licensing Manager

Document ID: 5014536

Attachments: As Stated

cc: Mr. Christopher Regan, USNRC (hardcopy letter and CD)  
HUG Licensing Correspondence Distribution (CD)  
Holtec Groups 1, 2, and 4 (w/o attach.)

**LAR 1014-2, REVISION 2**

**ATTACHMENT 1**

**SUMMARY OF PROPOSED CHANGES**

**LAR 1014-2, REVISION 2, SUPPLEMENT 25 SUMMARY OF PROPOSED HI-  
STORM 100 SYSTEM CHANGES**

**SECTION I – PROPOSED CHANGES TO CERTIFICATE OF COMPLIANCE 1014**

**Proposed Change No. 1**

Certificate of Compliance, Section 1.b and Appendix B, Section 3.2:

Remove the specific reference to BORAL<sup>®</sup> neutron poison material to allow the use of an alternate, equivalent neutron poison material, METAMIC<sup>®</sup>, as defined in the FSAR.

**Reason for Proposed Changes**

This change is proposed to allow flexibility in choosing the neutron absorber material used in the MPC basket. The neutron absorber material METAMIC<sup>®</sup> is proposed as an alternative to BORAL<sup>®</sup>. Because of the absence of interconnected porosities, the time required to dry a METAMIC<sup>®</sup>-equipped MPC is expected to be less compared to an MPC containing the rolled cermet class of neutron absorbers such as BORAL<sup>®</sup>.

**Justification for Proposed Changes**

METAMIC<sup>®</sup> neutron poison material has been demonstrated to be equivalent to BORAL<sup>®</sup> in performing the design function of absorbing thermal neutrons. METAMIC<sup>®</sup> is also equivalent to BORAL<sup>®</sup> in its thermal, structural, and shielding performance. The dimensions and tolerances for the fabrication and installation of the METAMIC<sup>®</sup> neutron absorber panels are identical to the current BORAL<sup>®</sup> dimensions and tolerances. The weight percent of B<sub>4</sub>C in METAMIC<sup>®</sup> is less than that for BORAL<sup>®</sup> given the same panel thickness, <sup>10</sup>B areal density, and <sup>10</sup>B loading penalty (25%).

METAMIC<sup>®</sup> has been considered in the criticality analyses in the same manner as BORAL<sup>®</sup> previously was considered, with one exception: only a 10% penalty on <sup>10</sup>B loading was considered for METAMIC<sup>®</sup> versus the previous licensing basis value of 25% for BORAL<sup>®</sup>. This change is appropriate because METAMIC<sup>®</sup> is essentially a solid material rather than a rolled cermet. Section 1.2.1.3 of Proposed FSAR Revision 2.B contains more detailed information regarding this change and Section 9.1 contains the qualification and production test program supporting the use of METAMIC<sup>®</sup> with 90% credit for <sup>10</sup>B.

EPRI Report 1003137, “Qualification of METAMIC<sup>®</sup> for Spent-Fuel Storage Applications” provides the pertinent qualification tests data for this material.

Holtec International proprietary Report HI-2022871, “Use of METAMIC<sup>®</sup> in Fuel Pool Applications”, includes a detailed discussion of the use of METAMIC<sup>®</sup> in wet storage applications, but also includes information germane to dry storage. Both of these reports support the conclusion that METAMIC<sup>®</sup> is well-suited for use in spent fuel storage casks. See proposed revisions to FSAR Sections 1.2.1.3, 4.2, 5.3, 6.4.11, and 9.1 in Attachment 5 for additional discussion.

Note: Appropriate conforming editorial changes to the MPC design drawings will be made after approval of the CoC amendment.

## **Proposed Change No. 2**

Certificate of Compliance, Section 1.b and 9; Appendix A, LCO 3.3.1; and Appendix B, Table 2.1-1:

- a. Modify CoC Section 1.b, Appendix A, LCO 3.3.1, and Appendix B, Section V of Table 2.1-1; and add new Section VIII to Appendix B, Table 2.1-1 to authorize damaged fuel for loading into the MPC-32 and damaged fuel and fuel debris for loading into the MPC-32F.
- b. Revise LCO 3.3.1 to re-format the required minimum soluble boron requirements for MPC-32/32F to provide the appropriate values for soluble boron based on fuel assembly array/class, intact vs. damaged fuel, and initial enrichment.

### **Reason for Proposed Changes**

- a. Damaged fuel and fuel debris currently are not authorized for loading in the MPC-32. Users currently must load PWR damaged fuel and fuel debris in the MPC-24E and -24EF. This change would enable customers to load all MPC-32 canisters on their ISFSI if they choose to do so.
- b. The reformatting of the MPC-32/32F soluble boron requirements reduces the current, across-the-board soluble boron concentration of 2,600 ppmb for MPC-32 to account for differences in fuel types and enrichments. This change can help reduce the amount of radioactive waste produced at a plant if the boron concentration in the spent fuel pool must be temporarily increased for cask loading.

### **Justification for Proposed Change**

- a. The addition of damaged fuel and fuel debris as authorized contents in the MPC-32 and MPC-32F has been analyzed and found to be acceptable.

The creation of MPC-32F entails only the thickening of the MPC shell at the top (with an associated reduction in the diameter of the MPC lid) and increasing the size of the lid-to-shell weld. This design difference is exclusively needed for qualification of the dual-purpose MPC for 10 CFR 71 transport loads - see proposed changes to FSAR Section 2.1.3 and new FSAR Figure 2.1.9. The rest of the MPC-32 and MPC-32F shell and basket designs are identical. This is the same design detail previously approved for the MPC-68F, MPC-68FF, and MPC-24EF in earlier CoC amendments. Allowing users to load damaged fuel and fuel debris into 32-assembly MPCs instead of 24-assembly MPCs reduces the risk of operating events and reduces the overall dose to personnel from ISFSI operations by reducing the total number of casks required to store a given amount of spent fuel. The MPC-32/32F damaged fuel container is shown in new FSAR Figure 2.1.2D. The technical evaluation is summarized below by discipline.

### Structural

The generic MPC-32/32F damaged fuel container (DFC) design is different in two respects from the previously approved generic MPC-24E PWR DFC: 1) the wall thickness is reduced from 0.075 inch to 0.0239 inch and 2) there is one additional spot weld per side in the MPC-32/32F DFC baseplate. A structural evaluation for the MPC-32/32F DFC is documented in the MPC structural calculation package (Report HI-2012787, Supplement 25) and all safety factors remain greater than 1.0.

### Thermal

The storage of DFCs containing damaged fuel assemblies in the peripheral fuel cells in MPC-32/32F is acceptable because, in a bounding evaluation, the effect of the presence of DFCs on *fuel basket thermal performance is calculated and per assembly decay heat loads for damaged assemblies reduced accordingly*. ~~peak fuel cladding temperature is negligible (i.e., much less than 1°F).~~

### Shielding

Generic damaged fuel has been analyzed in the MPC-24 and the MPC-68. The effect of storing damaged fuel and the post-accident consequences of collapsed damaged assemblies has been analyzed for the MPC-24 and the MPC-68. The results presented in the FSAR for the MPC-24 and the MPC-68 conclude that there is little effect on the external dose rates as a result of storing damaged fuel assemblies in these baskets.

Since storage of damaged fuel in the MPC-32 is similar to the MPC-24 and MPC-68 in that a limited number of assemblies are stored on the periphery of the basket, the effect on the external dose rates from storing damaged fuel in the MPC-32 will be similar to the effect seen in the MPC-24 and the MPC-68. Based on the results for the MPC-24 and MPC-68, it is concluded that the effect on the external dose rates from storing damaged fuel in the MPC-32 will be small. Therefore, storage of damaged fuel in the MPC-32 is acceptable from a shielding perspective without performing explicit MCNP calculations. Section 5.4.2 of the FSAR has been modified to add the above discussion pertaining to MPC-32.

### Criticality

Criticality evaluations were performed for the MPC-32/32F with intact fuel and damaged fuel/fuel debris using the same bounding fuel debris model developed in HI-STORM Amendment 1 for the MPC-68/68FF and the MPC-24E/EF. Additional calculations were performed to demonstrate that this model is conservative in the presence of soluble boron. Details of the damaged fuel model and calculations are discussed in general in FSAR Section 6.4.4.2, and calculations for the MPC-32/32F are specifically addressed in Section 6.4.4.2.6. These proposed FSAR changes may be found in Attachment 5.

Note that some of the reactivities reported in Tables 6.1.5 and 6.1.6 have increased slightly, although the corresponding soluble boron requirement were not changed. This is the result of a more extensive and slightly more conservative set of evaluations regarding the water density and the fill status of the guide tubes (see Tables 6.4.10, 6.4.11 and 6.4.14). These evaluations were necessary for consistency between the assembly classes and soluble boron levels.

### Confinement

There is not impact on the MPC enclosure vessel pressure boundary design or performance. Therefore, the MPC remains leak tight.

- b. The re-formatting of the minimum boron concentration is consistent with the supporting criticality evaluations. FSAR Section 6.4, and specifically Section 6.4.2.1.2, (Attachment 5) contains the details of the supporting evaluations. Users who previously may have had to increase the boron concentration in the spent fuel pool to load an MPC-32, may not need to do so if their normal spent fuel pool soluble boron concentration is sufficiently high. The eliminates the radioactive waste produced when

boron concentration is temporarily increased for cask loading and subsequently decreased for normal pool operation.

### **Proposed Change No. 3**

#### Certificate of Compliance, Sections 1.a and 1.b:

Revise the wording in these two CoC sections as follows:

- a. In Section 1.a and the first paragraph of Section 1.b, delete the “100 or 100S” designation in the references to the HI-STORM overpack
- b. In the second paragraph of Section 1.b, clarify that some early vintage MPCs include aluminum heat conduction elements (AHCEs).

Note: Conforming editorial changes to the affected MPC drawings will be made after approval of this amendment request.

- c. In the third paragraph of Section 1.b, delete all information pertaining to the authorized contents of each MPC model and add a statement defining the suffix to the MPC model number.
- d. In the fourth paragraph of Section 1.b, change the word “types” to “sizes” in two places in reference to the HI-TRAC transfer cask.
- e. In the fifth paragraph of Section 1.b: i) clarify the description of the differences between the HI-STORM 100S and HI-STORM 100 overpacks, ii) specify number of overpack air inlets and outlets as minimums, and iii) delete the terms “standard” and “short” from the discussion of HI-STORM 100A.

#### **Reason and Justification for Proposed Changes**

- a. These changes for the overpack description are proposed for consistency with the discussion of the HI-TRAC transfer cask and MPCs in these portions of the CoC.
- b. For those MPCs loaded under CoC Amendment 2 or later, the AHCEs are prohibited because they have not been included in the thermal evaluation model. In the thermal evaluation for those MPCs loaded under the original CoC or Amendment 1, the aluminum heat conduction elements were conservatively modeled as a flow restriction, but no credit was taken for heat transfer through them in the bounding thermal analysis presented



in FSAR Revision 1; therefore, the AHCEs are optional equipment for MPCs loaded under the original CoC or Amendment 1. There are a number of MPCs that are, or will be loaded under the original CoC or Amendment 1 that contain AHCEs. Therefore, this proposed change is consistent with past and future MPCs and the supporting thermal analyses. Sections 1.2.1.1 and 4.4.1.1.b of the proposed FSAR (Attachment 5) have been modified appropriately to address this change. See also Proposed Change No. 22.

- c. This information currently duplicates Section 6 of the CoC, which refers to Appendix B of the CoC for approved contents. Appendix B of the CoC contains detailed specifications for the contents of each MPC model, including all of the information contained in the material proposed for deletion. This change eliminates redundancy in the CoC.
- d. This wording change provides clarification in distinguishing between the 125-ton and the 100-ton HI-TRAC transfer casks. The term “types” is too general and subject to misinterpretation. The term “sizes” is more correct for distinguishing between the 100-ton and 125-ton transfer casks.
- e. These wording changes provide i) clarification of the major differences between the 100S and 100 overpack designs, ii) flexibility regarding the number of air inlets and outlets for potential future modifications, and iii) clarification by removing redundant terms for the HI-STORM 100 and HI-STORM 100S overpack designs.

### **Proposed Change No. 3a**

This proposed change has been superseded by Proposed Change No. 22 in Revision 1 to this LAR. See Section ~~II/V~~ of this document.

### **Proposed Change No. 4**

Certificate of Compliance, Appendix A, SR 3.1.1.1, SR 3.1.1.3 and Table 3-1:

- a. Revise Surveillance Requirement (SR) 3.1.1.1 and Table 3-1, and relocate information previously in Table 3-1 to new Table 3-2 as shown in the attached markup CoC to reflect necessary changes in requirements for MPC cavity drying.
- b. Revise SR 3.1.1.3 to remove the helium leakage test requirement.

- c. ~~Revise~~*Relocate* the helium backfill requirements in new Table 3-2 (previously located in Table 3-1) as shown in the attached mark-up of the CoC.

### **Reason for Proposed Changes**

- a. These proposed changes in MPC cavity drying requirements are necessary as a consequence of higher authorized heat loads and the new peak fuel cladding temperature limit suggested by ISG-11, Revision 3~~2~~<sup>1</sup>. There are now a variety of requirements and options based on the ~~decay heat load of the MPC and the~~ burnup of fuel being stored (i.e., moderate burnup versus high burnup fuel).
- b. This reflects the designation of the MPC as leak tight in accordance with the guidance in ISG-18.
- c. This proposed change is a result of lessons learned in the field on implementing the current helium backfill pressure requirement of 29.3 to 33.3 psig. Due to the accuracy of instruments available for performing this activity in then field, more precision was required in establishing the appropriate range in the Technical Specifications.

### **Justification for Proposed Changes**

- a. The proposed changes in MPC cavity drying requirements create the necessary controls to ensure the peak fuel cladding temperature limit of 400°C is not exceeded during short term loading operations *with high burnup fuel assemblies*. They also provide ~~optional requirements (fuel cladding hoop stress calculations)~~ for MPCs containing all moderate burnup fuel ( $\leq 45,000$  MWD/MTU) to allow the fuel cladding temperature to approach ~~the previous licensing basis limit of 570°C during vacuum drying, for heat loads up to those already licensed in Amendment 1 to the CoC.~~ Any MPC containing one or more high burnup fuel assemblies must be dried using the forced helium dehydration method, in which case the 400°C temperature limit is ensured for all authorized heat loads. See proposed changes to FSAR Section 4.5 in Attachment 5 ~~and Holtec calculation HI-2033054, being provided under separate cover,~~ for details of the thermal analyses.
- b. Human factors improvement.

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<sup>1</sup> The modified MPC cavity drying requirements also reflect the fuel cladding hoop stress calculation option to retain the 570°C temperature limit for moderate burnup fuel ( $\leq 45,000$  MWD/MTU). This is expected to be consistent with the soon-to-be-published ISG-11, Revision 3.

- c. The TS requirements for helium backfill more accurately account for the potential range of instrument accuracies in the field, the different MPC cavity drying methods, and the supporting thermal analyses. The thermal analyses evaluate a lower bound helium backfill value that ensures a sufficient density of helium is in the MPC to promote adequate thermosiphon heat transfer. They also evaluate an upper bound value to ensure the MPC design pressure is not exceeded. See proposed changes to FSAR Section 4.4.1 in Attachment 5 for additional justification.

### **Proposed Change No. 5**

#### **Certificate of Compliance, Appendix A, LCO 3.1.3 and associated Bases in FSAR Appendix 12.A:**

Revise this LCO and associated Technical Specification Bases as shown in the attached markup of the CoC and FSAR Appendix 12.A to:

- a. Provide appropriate requirements for ensuring MPC cavity bulk helium temperature is less than 200 degrees F prior to re-flooding, instead of the existing “helium gas exit temperature.” Revise associated bases in the FSAR accordingly.
- b. Change the Completion time of Required Action A.2 from 22 hours to “Immediately.”

#### **Reason for Proposed Changes**

- a. Using a forced helium recirculation system to cool the MPC cavity gas for low decay heat load casks may be unnecessary in the unlikely event that an MPC must be unloaded. This change provides appropriate flexibility for users who may have to unload an MPC with a low decay heat load.
- b. This change is required as a result of the new, lower peak fuel cladding temperature limit of 400°C during short-term operating conditions, including unloading operations.

#### **Justification for Proposed Change**

- a. Depending upon the decay heat in the cask at the time of unloading, it may not be necessary to cool the contained helium with a recirculating helium cooldown system prior to re-flooding with water. The helium temperature of very low decay heat load casks could be less than 200 degrees F at the time of re-flooding with no action required. Alternatively, adequate

cooling of the helium inside the MPC may be able to be accomplished by non-intrusive means, such as air or water applied to the outside surface of the MPC.

The bulk temperature of the helium in the MPC and the fuel cladding itself is predicted using a computational fluid dynamics computer program (FLUENT, FSAR Chapter 4) to license the cask for normal, off-normal, and accident conditions of storage. These same analytical techniques, accepted as the basis for loading operations and long-term fuel storage, can be used to predict the bulk helium temperature of an MPC designated for unloading. The actual characteristics of the MPC contents (i.e., fuel type, presence of non-fuel hardware, time in storage) can be used to conservatively predict the bulk helium temperature prior to re-flooding. The results of that prediction would then be used to determine the appropriate means (if any are necessary) and time frame to cool the bulk helium down to 200°F prior to re-flooding in order to minimize thermal stress in the fuel cladding.

- b. The thermal analyses described in FSAR Section 4.5 indicate that, *for MPCs containing high burnup fuel assemblies, there are threshold decay heat loads below which MPCs may be emplaced in the HI-TRAC transfer cask without supplemental cooling. Above these threshold decay heat loads, supplemental cooling is required while in a HI-TRAC transfer cask. FSAR Section 4.5 addresses specific examples of supplemental cooling. However, the particular type of augmented cooling is necessarily site-specific and is left to the user to determine, using the thermal methodologies in the HI-STORM FSAR.*

### **Proposed Change No. 5a**

Certificate of Compliance, Appendix A, LCO 3.1.4 (new) and associated Bases in FSAR Appendix 12.A:

New LCO 3.1.4, “Supplemental Cooling System,” is proposed to be added to the CoC and associated Bases for this LCO added to the FSAR, both as shown in the attached markups to these documents.

### **Reason and Justification for Proposed Change**

Second round RAI number 11-1 requested a new LCO for the proposed HI-TRAC cooling system necessary, for *high burnup fuel assemblies certain heat loads*, to maintain fuel cladding temperatures below applicable limits during normal onsite transfer operations. The LCO, Required Actions, and Surveillance Requirements

are based on the supporting thermal analyses of the system as discussed in the proposed new Bases for the LCO.

### **Proposed Change No. 6**

Certificate of Compliance, Appendix A, LCOs 3.2.1 and 3.2.3; Action B.1 of LCO 3.1.2; and Section 5.0:

Delete LCOs 3.2.1 and 3.2.3 and associated bases in FSAR Appendix 12.A and replace them with new Technical Specification Program 5.7 for radiation protection, located in CoC Appendix A, Section 5.0. Modify the Required Action in LCO 3.1.2 to conform with this change.

#### **Reason for Proposed Change**

The current Required Actions for LCOs 3.2.1 and 3.2.3 do not lead to an end point that results in compliance with the LCO requirements. For example, if dose rates on the HI-TRAC transfer cask exceed one of the LCO 3.2.1 limits, Required Actions A.1 and A.2 of that LCO require the cask user to administratively verify correct fuel loading and to perform an evaluation to verify compliance with 10 CFR 20 and 10 CFR 72, respectively. Once these actions are complete, operations are permitted to continue, yet the cask surface dose rates would remain out of compliance with the LCO limits. The same logic applies to LCO 3.2.3 for HI-STORM overpack dose rates.

In addition, this change is proposed to be consistent with the guidance of NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance" and with many plants' Part 50 technical specifications. General licensees' radiation protection programs that implement the ALARA philosophy are considered sufficient to protect operations personnel and the public and to ensure compliance with regulatory dose limits.

The program description also includes specific requirements on determining site-specific cask contact dose rate limits based on the licensee's evaluation performed pursuant to 10 CFR 72.212. These additional requirements provide continued NRC control over certain aspects of any shielding analyses and evaluations performed to demonstrate compliance with off-site dose limits and in support of changes made under the provisions of 10 CFR 72.48.

#### **Justification for Proposed Change**

There are no numerical regulatory limits on contact dose rates from a spent fuel storage cask in 10 CFR 72 or 10 CFR 20. For normal and off-normal operations, however, general licensees must demonstrate compliance with 10 CFR 72.104 for

dose at the controlled area boundary. Compliance with 10 CFR 72.104 is site-specific, based on the dose rate from reactor operations, contents of the casks, the number of casks at the ISFSI, local meteorology, and the distance to the site boundary. The contact dose rate limits in current LCOs 3.2.1 and 3.2.3 serve no function for licensees in demonstrating compliance with 10 CFR 72.104.

Compliance with surface dose rate limits in the Technical Specifications (TS), or elsewhere, are not a reliable indicator of proper cask loading or consistency with the site-specific off-site dose analysis. Specifically, if a measured cask surface dose rate exceeds the cask TS value, certainly a mis-loading has occurred. However, measuring a surface dose rate less than the limit in no way assures that all contents loaded meet the CoC requirements. This is because the actual contents of a cask loaded at a given general licensee's facility will never match the bounding design basis contents used in the licensing basis shielding analyses. Individual fuel assemblies or non-fuel hardware not meeting the CoC could be loaded with the overall effect on dose rate being insignificant. The administrative controls used to select and document fuel assemblies and non-fuel hardware chosen for loading in a cask (equivalent to those used to store fuel in a plant's spent fuel pool) are the only reliable way to ensure the fuel loading requirements of the CoC are met.

Contact dose rates from the casks are a factor in determining occupational exposures during cask loading operations. Occupational exposure regulatory limits are set by 10 CFR 20 and exposures to personnel are generally controlled to even lower limits through the users' ALARA-based radiation protection programs. The dose rates to personnel from a loaded HI-STORM overpack or HI-TRAC transfer cask are necessarily site-specific, and cask specific, based on the particular contents of the cask. Part 50 licensees are well-versed at handling radioactive containers, many of which emit much higher levels of radiation than a dry storage cask. Therefore, these requirements are more appropriately controlled through a Technical Specification program.

See also the response to Round 1 RAI Question 10-3.

Note: In response to second round RAI Number 10-1, limiting contact dose rate on the top and sides of the overpack have been added.

### **Proposed Change No. 7**

Certificate of Compliance, Appendix B, Section 1.0, Definitions; Table 2.1-1, Note 1 in Sections I, IV, V, VII, and VIII; and Note 3 of Table 2.1-8

Revise the definition of NON-FUEL HARDWARE as shown in the attached mark-up of the CoC to include vibration suppressor inserts. Revise the subject

notes as shown to allow the storage of vibration suppressor inserts as integral non-fuel hardware that may be stored in the MPC with a fuel assembly.

### **Reason for Proposed Change**

Vibration suppressor inserts have been identified by a number of Holtec's clients as non-fuel hardware that is integral to the fuel assemblies and must be qualified for storage. Vibration suppressor inserts were added by certain fuel vendors as a design feature to address a vibration-induced failure problem in operating reactors.

### **Justification for Proposed Change**

The vibration suppressor inserts contain no fissile material and have been evaluated as activated hardware (BPRAs). See Section 5.2.4 of the proposed FSAR changes (Attachment 5) for additional information. Table 2.1-8 of CoC Appendix B has been modified to include the vibration suppressor inserts with the existing approved fuel insert burnups and cooling times.

## **Proposed Change No. 8**

Certificate of Compliance, Appendix A, LCO 3.3.1; Appendix B, Table 2.1-1, Section IV; and Appendix B, Table 2.1-2;

Increase the maximum authorized initial enrichment for PWR damaged fuel and fuel debris to 5.0 wt.% as shown in the attached mark-ups of the CoC

### **Reason for Proposed Change**

PWR users have damaged fuel and fuel debris up to 5 wt.% initial enrichment that needs to be placed into dry storage.

### **Justification for Proposed Change**

Damaged fuel and fuel debris up to 5.0 wt.% <sup>235</sup>U has been evaluated and found to be acceptable for loading in the PWR MPCs. See Sections 6.4.4.2.5 and 6.4.4.2.6 in the attached proposed FSAR changes for detailed justification.

## **Proposed Change No. 9**

Deleted

## **Proposed Change No. 10**

### Certificate of Compliance, Appendix B, new proposed Section 2.3:

Provide a process for the certificate holder to request and receive NRC approval of case-specific alternatives to the cask contents on behalf of a cask user, as shown in the attached markup of the CoC.

### **Reason for Proposed Change**

To provide necessary flexibility for the NRC to review and approve, upon request by Holtec, small deviations from the cask contents limits in the CoC that have been shown to have little or no safety significance. This change process will eliminate the need for licensees to request exemptions from the regulations or significantly delay their fuel loading schedules for small, non-safety significant changes to the CoC cask contents on a case-specific basis.

### **Justification for Proposed Change**

This proposed change is consistent with NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance.” NUREG-1745 also suggests moving some of the cask contents limits to the FSAR. However, Holtec has chosen to leave all cask contents parameters the same as currently found in the CoC (although some of the values for those parameters are proposed to be changed in this amendment request). This is conservative since a CoC amendment would still be necessary to permanently change any of the parameters (or the values) in the CoC, including any case-specific changes approved by the NRC under this process. The flexibility permitted by this proposed change is appropriate because there may be instances where cask users’ fuel or other contents have slight deviations from the limits in the CoC, where there is little or no safety concern with granting the deviation on a case-specific basis.

This change process allows Holtec to support our customers’ fuel loading schedules without the users having to request exemptions from the regulations for small deviation from the approved contents section of the CoC. Holtec will then pursue a permanent change to the parameter or value on a normal priority schedule, using the CoC amendment process. Overall, this change process eliminates unnecessary regulatory burden in an area of little or no safety significance, but retains the requirement for prior NRC approval of cask contents changes.



## **Proposed Change No. 11**

Certificate of Compliance, Appendix A, LCO 3.1.2 - Required Actions B.2.1 and B.2.2, and SR 3.1.2.1; and Appendix B, Tables 2.1-1 and 2.1-4 through 2.1-7:

Revise Table 2.1-1, delete Tables 2.1-4 through 2.1-7, and create new Section 2.4 in Appendix B as shown in the attached markup of the CoC to provide new (higher) limits for fuel assembly decay heat, and for burnup as a function of decay heat, enrichment, cooling time, and fuel array/class. Modify the Completion Times for Required Actions B 3.2.1 and B 3.2.2. ~~to reflect the revised blocked duct accident analysis. Modify the acceptance criterion for temperature measurement in SR 3.1.2.1 to be 145°F to conform to these changes.~~ See also Proposed Change 15a.

### **Reason for Proposed Changes**

Based on user input, the existing limits unnecessarily penalize certain fuel types due to only grouping by reactor type (PWR or BWR). The previous limits did not meet the entire spectrum of users' needs to store fuel with higher heat emission rates. ~~Other changes are conforming changes made necessary by the higher heat loads.~~

### **Justification for Proposed Change**

#### Thermal

The previous burnup and decay heat limits were distinguished only by PWR or BWR fuel type for each MPC model. The revised limits are specified by fuel array/class and MPC model to provide an improved specificity for the various fuel types. ~~The new limits appropriately reflect the ability of the HI-STORM 100 System to reject more heat than previously authorized, while still retaining adequate margins to the various limits (see revised FSAR Section 4.4 in Attachment 5).~~ Placing the higher burnup fuel in the central core of the basket, surrounded by lower burnup fuel reduces the overall dose to personnel and the public from ISFSI operations due to the self-shielding phenomenon of the fuel assemblies. ~~See proposed revisions to FSAR Section 4.4.1.1.9 (Attachment 5) for additional justification.~~ The permissible fuel cladding temperature limit used to determine the maximum cask heat loads are consistent with ISG-11, Revision 2. (see also Proposed Change Number 15a).

#### Shielding

The shielding analysis in Chapter 5 of the FSAR has been modified to reflect the changes in the allowable burnup and cooling times by changing all dose rate

calculations using the design basis fuel assemblies, B&W15x15 and GE7x7. The source terms have also been changed appropriately. The choice of design basis fuel assembly for the shielding analysis remains the same. Section 5.2 has been modified slightly to address the fact that the different array classes have different burnup and cooling times as a result of this change. The design basis assemblies remain valid because the analysis in Chapter 5 uses the maximum burnup from all array classes for a given cooling time. This is described in Section 5.1 of the proposed Revision 2 FSAR (Attachment 5).

In conjunction with calculating the allowable burnups for the different array classes, Tables 5.2.25 and 5.2.26 have been slightly modified. In Table 5.2.25, the pellet diameter and resulting uranium loadings of three of the assemblies have been increased to be consistent with the maximum permissible value in the CoC. In Table 5.2.26, the 9x9 assembly has been modified to reflect the 9x9 array class which now has the highest decay heat load for the specified burnup and cooling time in that table.

In the calculation of the allowable burnups for the different array/classes an additional change was made in the shielding analysis. Rather than use the same power level of 40 MW/MTU for all array/classes, the power per assembly was calculated for each reactor type and increased by 10 or 20% to account for potential power uprates for the PWR and BWR plants, respectively. Tables 5.2.25 and 5.2.26 reflect this change as does Section 5.2.5 in Attachment 5.

### Accidents

Placing the relatively hotter fuel assemblies in the center of the MPC basket by design obviates the need to analyze a fuel assembly mis-loading accident. This is because, as described in FSAR Section 4.4.1.1.9, the inadvertent loading of a hotter fuel assembly on the periphery is actually more advantageous from a thermal perspective (i.e., the heat rejection of the cask system would be better with hotter assemblies on the periphery of the fuel basket). From a shielding perspective, the mis-loading of an assembly would result in a small, localized increase in the contact dose rate on the cask and would be imperceptible at the controlled area boundary.

~~The 100% air duct blockage accident was re-analyzed for the design basis heat loads to yield two heat load dependent Completion Times in the TS. This re-analysis is discussed in FSAR Section 11.2.13 (Attachment 5). The results of the Amendment 1 analyses show that, for heat loads  $\leq 27.74$  kW (the Amendment 1 maximum heat load), no components reach their short term temperature limit over the 72 hour duration of the analysis. For a bounding MPC 68 heat load of 35.5 kW in Amendment 2, no components reach their short term temperature limit for 24 hours. The Completion Times for Required Actions B.2.1 and B.2.2 of this~~

~~LCO have been revised to reflect these results.~~—Note also that the basis for the ~~revised~~ Completion Times no longer includes the assumption that the complete blockage of all inlet ducts occurs immediately after completion of the last surveillance. This change is consistent with the bases for Completion Times in power reactor technical specifications, which are developed assuming that the degraded condition begins at the time the component or system is declared inoperable<sup>2</sup>. It is not required to assume the component or system has been inoperable since the last successful completion of the Surveillance Requirement. See also the Bases for LCO 3.1.2 in FSAR Appendix 12.A (Attachment 5).

### **Proposed Change No. 12**

Certificate of Compliance, Appendix B, Tables 2.1-2 and 2.1-3:

Revise the maximum allowable uranium masses for certain fuel assemblies as shown in the attached markup to the CoC. These changes are made to maintain consistency with the revised shielding analyses in Chapter 5.

#### **Reason for Proposed Changes**

As Proposed Change 11 discusses, the allowable burnups are being calculated in this LAR for different array classes rather than a single PWR or BWR array/class. Explicit analysis has been performed to determine the allowable burnups for each array/class. The change to the allowable uranium mass loadings is being made to reflect the actual uranium mass loadings used in the calculation of the allowable burnups for each array/class.

#### **Justification for Proposed Change**

The allowable burnups as a function of cooling time are calculated using the allowable decay heat as an input. Source term calculations are performed to determine the burnup that produces the allowable decay heat for a specified cooling time. One of the key inputs in the source term calculations is the uranium mass loading. The maximum uranium mass loading has historically been specified in the CoC for the HI-STORM system and has always been a quantity derived from the shielding analysis. Therefore, this change is being made to maintain the uranium mass loading values consistent with the shielding analysis used to determine the allowable burnups. The maximum uranium mass loadings in the CoC are not based on the criticality analysis or the thermal analysis and

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<sup>2</sup> This is not to say the actual point of inoperability is not an issue to be investigated through the root cause evaluation conducted in accordance with the corrective action program, if necessary.

changes to these mass values do not reflect changes in the criticality or thermal areas.

### **Proposed Change No. 13**

Certificate of Compliance, Appendix B, Table 2.1-8:

Revise the maximum allowable burnup for non-fuel hardware inserts as shown in the attached markup to the CoC. These changes are made to maintain consistency with the revised shielding analyses in Chapter 5.

#### **Reason for Proposed Changes**

Feedback from our clients has indicated that the allowable burnups versus cooling time for some of the non-fuel hardware is unnecessarily restrictive. Therefore, the allowable burnups for a given cooling time are proposed to be increased for non-fuel hardware inserts.

#### **Justification for Proposed Change**

The allowable burnups for the non-fuel hardware are derived from the shielding analysis where a maximum activity of Cobalt-60 is specified for the non-fuel hardware and the burnups are chosen at a given cooling time to assure that the calculated Cobalt-60 activity remains less than the maximum value used in the shielding evaluation. In order to increase the burnups for the non-fuel hardware inserts, the maximum permissible Cobalt-60 activity was increased as identified in Section 5.2.4 and Table 5.2.31 (Attachment 5). The dose rates reported in Chapter 5 of the FSAR were modified to account for this increase in source term.

### **Proposed Change No. 14**

Certificate of Compliance, Appendix B, Section 3.3 and Table 3-1:

- a. Change “Exceptions” to “Alternatives” throughout the section.
- b. Revise Section 3.3 as shown in the attached mark-up of the CoC to clarify the ASME Code Edition of record for the HI-STORM 100 System. This clarification is proposed to allow the latest effective versions of ASME Sections V and IX to govern the performance of non-destructive examination (NDE) and welding, respectively.

- c. Add the new and revised ASME Code alternatives as shown in the attached markup of the CoC.
- d. Add “on a case-specific basis” to the requirements related to alternatives to the Code as shown in the attached markup of the CoC.
- e. In the second paragraph of the justification for the alternative to Code Article NB-6111, change “process” to “results” and add “relevant” before “findings.”

### **Reason and Justification for Proposed Changes**

- a. This is an editorial change to make the CoC agree with the regulatory guidance terminology contained in ISG-10.
- b. Code Sections V and IX are periodically revised by the ASME to more closely reflect the state of the art in NDE and welding. It is prudent to require the performance of these activities to be in accordance with the latest techniques endorsed by ASME. This change does not affect the design or analysis of the storage system in any manner and is consistent with the current practice of the fabricator of the components governed by the Code. A requirement has been added to the FSAR and CoC for the certificate holder to perform a Code reconciliation prior to the fabricator’s adoption of a later edition of Section V or IX.
- c. These new and revised Code alternatives are needed to reflect the design drawings and are identical to those previously reviewed and approved by the NRC under separate cover for certain serial number cask components. See NRC letters to Holtec dated March 5<sup>th</sup> and March 7<sup>th</sup>, 2002.
- d. Based on ISG-10, the intent of this flexibility is to authorize additional Code alternatives on a case-specific basis. This change provides the necessary clarification.
- e. This is an editorial change to make the CoC agree with the regulatory guidance terminology contained in ISG-4.

### **Proposed Change No. 15**

Certificate of Compliance, Appendix B, Section 3.5:

Revise the text in Appendix B, Section 3.5.1 as shown in the attached markup to the CoC. These changes are made to maintain consistency with similar wording in CoC Appendix A, Section 5.5.

#### **Reason and Justification Proposed Change**

This proposed change clarifies the text to state that the Cask Transfer Facility design criteria requirements do not apply to lifting devices integral to structures governed by the regulation of 10 CFR 50. Our users have stated that the use of the word “outside” as currently written in Section 3.5.1 could be misconstrued to mean anywhere “outdoors”, which could include outdoor cranes integral to the Part 50 facility and governed by Part 50 regulatory requirements. This is not the intent of this CoC requirement. The intent of the requirement is to distinguish between 10 CFR Part 50 and Part 72 jurisdiction.

### **Proposed Change No. 15a**

Certificate of Compliance, Appendix A, LCOs 3.1.1, 3.1.2, 3.1.3, and Section 5.6; and Appendix B, Section 1.0, Section 2.1.2, Section 2.1.3, Tables 2.1-1 through 2.1-3; and Tables 2.4-4 through 2.4-8, and new Section 3.4.10:

Revise the affected portions of the referenced CoC sections and tables to reflect incorporation of the review guidance contained in ISG-11.

#### **Reason and Justification Proposed Changes**

These changes reflect the necessary modifications to the CoC to adopt the revised review guidance contained in ISG-11, Revision 2. The changes to the CoC are consistent with the guidance with regard to the authorization for storage of any fuel cladding material previously approved by NRR for use in a commercial reactor, the elimination of fuel cladding oxidation thickness as a criterion for classifying fuel as damaged, the elimination of the confinement source term penalty factors for high burnup fuel, and the use of a single temperature limit for long-term storage and short-term operating conditions (e.g., vacuum drying). The supporting justification for these changes is contained in proposed changes to FSAR Chapter 4, with conforming changes in FSAR Chapters 1, 2, 5, 8, 10 and 12, and in the thermal-hydraulic calculation package, HI-2033054.

## SECTION II – PROPOSED CHANGES TO THE FSAR

### **Proposed Change No. 16**

#### Changes to FSAR Chapter 2, Tables 2.2.1 and 2.2.3:

- a. Revise the off-normal MPC internal design pressure from 100 psig to 110 psig as shown in FSAR Table 2.2.1 (Attachment 5).
- b. Revise the normal temperature limit for the overpack lid top plate as shown in FSAR Table 2.2.3 from 350°F to 450°F in the attached proposed FSAR markups (Attachment 5).

#### **Reason for Proposed Change**

- a and b. Increasing these pressure and temperature limits *provides the greatest assurance that structural analyses performed in FSAR Chapter 3 bound the current and potential future operating conditions.* ~~is necessary to support the increased thermal loads being proposed elsewhere in this amendment request.~~

#### **Justification for Proposed Change**

- a. The off-normal condition is defined as the upset or Level B condition in the ASME Code for which the allowable stresses are 10% greater than for normal service conditions. Therefore, the associated permissible pressure may be increased by 10%.
- b. The higher lid top plate temperature has been evaluated and found to be acceptable. The overpack lid lifting evaluations contained in FSAR Section 3.4.3.7 address the adequacy of the threaded holes used to lift the lid for placement on the loaded overpack. This evolution occurs with the lid at ambient temperature. Therefore, this evaluation is unaffected by this change.

The change in the design lid material temperature does not affect the vertical drop or tipover analyses.

The missile impact on the top lid was re-evaluated for the increased lid plate design temperature. The allowable stress is slightly reduced and the safety factor is reduced accordingly, but still shows a safety margin of 33 percent. See proposed FSAR Section 3.4.8.1 (Attachment 5) for more detailed discussion of this event.

### **Proposed Change No. 17**

#### Changes to FSAR Chapter 3 and Chapter 7

Delete Appendices 3.B thru 3.AS in their entirety and re-locate this information to the supporting calculation package. Create new FSAR Subsections 3.4.4.3.1.8 and 3.4.4.3.1.9 to address some of these calculation results.

#### **Reason and Justification for Proposed Change**

These detailed calculations are of a level of detail that is not appropriate for the FSAR. The supporting calculation packages have been updated as necessary to include the appropriate information deleted from the FSAR appendices.

### **Proposed Change No. 18**

This proposed change is deleted in light of the issuance of ISG-18 since the original submittal of this LAR. Please see the response to first round RAI Question 7-1.

### **Proposed Change No. 19**

#### Change to FSAR Chapter 11

In Section 11.1.4.3, remove discussion of the three-ducts blocked condition. Remove results currently presented in Table 11.1.2

#### **Reason and Justification for Proposed Change**

The three-ducts blocked condition was previously included in the FSAR for comparison purposes only. This comparison is now being removed. The design basis off-normal condition is two ducts blocked and the design basis accident condition (FSAR Section 11.2.13) is all ducts blocked.

### **Proposed Change No. 20**

#### Changes to FSAR Chapter 13

Replace the detailed discussion of the Holtec QA program throughout Chapter 13 with a short discussion of the program and a reference to the current NRC-approved QA program in Section 13.0 (see Attachment 5). Sections 13.1 through 13.3 and 13.5 are deleted in their entirety. Section 13.4 and Appendices 13.A and



13.B were removed in FSAR Revision 1 after Revision 13 of the Holtec QA Program Manual was approved by the NRC.

**Reason for Proposed Change**

To remove redundant information.

**Justification for Proposed Change**

The NRC has approved Revision 13 of Holtec's QA program under 10 CFR 71 (Approval 71-0784, Rev. 3). Holtec also uses this QA program to control activities important to safety that are performed under 10 CFR 72 as permitted by 10 CFR 72.140(d). Including the same, or similar QA program information in FSAR Chapter 13 is unnecessarily redundant. This change is similar to that approved for other Part 72 general certifications (e.g., Fuel Solutions, Docket 72-1026). In accordance with 10 CFR 72.140(d) The Holtec QA program also meets the supplemental recordkeeping requirements of 10 CFR 72.174 for use under Part 72.

## **SECTION I/IV – NEW CHANGES WITH FIRST AND SECOND ROUND RAI RESPONSES**

### **Proposed Change No. 21**

#### Certificate of Compliance, Various Locations

Make the following editorial changes to the CoC:

- a. On the first page of the CoC, delete “Inc.” from Holtec International’s name.
- b. In the third paragraph of Section 1.b, change the last word from “dimensions” to “diameter.”
- c. Condition 5: In the second sentence, change “safety” to “regulatory.”
- d. Appendix B, Table 2.1-1: In each of the sections for the PWR MPCs (MPC-24, -24E, -24EF, -32, and -32F), add or modify an item at the end of the section that clarifies that neutron sources are not permitted for storage.
- e. Appendix B, Table 2.1-1, Section VI.A.1.h: Delete “and DFC” from the fuel assembly weight entries (two places).
- f. Appendix B, Table 2.1-1, Sections VII and VIII: In Note B, change “MPC-24E” and “MPC-32” to “MPC-24EF” and “MPC-32F,” respectively.
- g. Appendix B, Table 2.4-3, Note 2: Replace “NON-FUEL HARDWARE” with “channels.”

#### **Reason and Justification for Proposed Changes**

- a. Editorial. The legal name of Holtec is “Holtec International, a New Jersey Corporation.”
- b. Using the term “dimensions” unnecessarily restricts the ability to potentially change the height of the MPC and other cask components under the provisions of 10 CFR 72.48, if the need arises. We believe that the diameter is the only dimension that should be controlled via CoC amendment.

- c. The reviews referred to in this sentence are to be conducted pursuant to 10 CFR 50.59 or 10 CFR 72.48. These reviews are regulatory reviews to ascertain whether prior NRC approval is required before the activity can be implemented. This is not to be confused with the evaluation of the safety of the activity, which is conducted under the appropriate quality assurance process (e.g., design control).
- d. This is a clarification. No PWR neutron sources have been certified for storage in the HI-STORM 100 System.
- e. Item VI.A.1 addresses storage of intact BWR fuel in MPC-68FF. DFCs are not required for intact fuel storage.
- f. Editorial
- g. “NON-FUEL HARDWARE” is a defined term in Section 1.0 of Appendix B for PWR fuel inserts. The term does not apply to the BWR MPC-68/68FF. “Channels” is the appropriate term.

### **Proposed Change No. 22**

#### Certificate of Compliance, Condition 11:

Modify the language in CoC Condition 11 as shown in the attached marked up CoC to address component certification and use. This change also prompted a conforming change to Section 1.b in the discussion pertaining to the aluminum heat conduction elements.

#### **Reason for Proposed Change**

This change is requested to clarify the intent of this CoC condition as it relates to amended CoCs and hardware certified to different CoC amendments.

#### **Justification for Proposed Change**

Over time, licensee users of the HI-STORM 100 System may receive licensed hardware components (MPC, overpack, and transfer cask) fabricated and certified to any of the approved amendments to the CoC. Unless specifically prohibited by the CoC, any component certified to any CoC amendment may be used with any other component certified to any amendment of the CoC, provided the CoC holder has confirmed the design compatibility of each licensed component for the applicable CoC amendment. For example, licensees receive one HI-TRAC transfer cask, which would have been certified to the CoC amendment effective at

the time of fabrication. Unless specifically prohibited by the CoC, that HI-TRAC transfer cask may be authorized for use under any later amendment of the CoC provided the CoC holder has performed the design compatibility assessment and certified to the licensee that this compatibility exists. This change is necessary to address a future potential concern with configuration control and component compatibility if, for example, an MPC is transported to the federal repository and the licensee wishes to re-use the “old” overpack in which that MPC was previously stored.

### **Proposed Change No. 23**

#### Certificate of Compliance, Appendix A, Table 3-1 and Appendix B, Section 3.6.2.4

Modify the MPC drying acceptance criterion applicable to the use of the Forced Helium Dehydration (FHD) System to include an alternative measurement of gas dew point exiting the MPC to confirm a partial water vapor pressure of 3 torr or less in the MPC.

#### **Reason for Proposed Change**

As part of Holtec’s prototype deployment of the FHD system at the Trojan Plant site, it was determined that an alternative, more direct measurement of the gas condition exiting the MPC was appropriate to consider.

#### **Justification for Proposed Change**

A dew point of the gas exiting the MPC of  $\leq 22.9^{\circ}\text{F}$  for  $\geq 30$  minutes corresponds to a partial water vapor pressure of 3 torr, which is the accepted dryness limit for spent fuel storage casks per NUREG-1536, Section 8.V.1.

### **Proposed Change No. 24**

#### Certificate of Compliance, Appendix B, Table 3-1, “List of ASME Code Alternatives”:

- a. In the “alternative, justification, & compensatory measures” column for Code Article NB-6111, replace the word “hydrostatically” and “hydrostatic” with “pressure” (two places).
- b. In the “alternative, justification, & compensatory measures” column for Code Articles NF-3256 and NF-3266, remove the term “by an ‘\*’.”

### **Reason and Justification for Proposed Changes**

- a. ASME Section III, Subsection NB, Article NB-6110 requires a pressure test of the vessel. The pressure test may be a hydrotest or, provided certain criteria are met, a pneumatic test. This change is proposed to allow users the option to use the flexibility that the Code already offers for pressure testing vessels. A conforming change to FSAR Table 2.2.15 (ASME Code Alternatives) is also proposed in support of this CoC change. It is not a change to the Code alternative itself, since the Code already allows either type of pressure testing.
- b. The type of notation used on the design drawings to indicate “non-NF” welds is not germane to the justification or compensatory measures associated with this Code alternative.

### **Proposed Change No. 25**

#### CoC Appendix B, Section 3.2.6

Modify the language in this CoC section as shown in the attached marked-up CoC to remove specific reference to fuel spacers.

### **Reason and Justification for Proposed Change**

The intent of the requirement is to ensure the active fuel region of the fuel assemblies is positioned within the neutron absorber region of the fuel storage cell. The method by which this accomplished should be left to the designer.

### **Proposed Change No. 25a**

#### CoC Appendix B, Section 3.2.7:

Revise Section 3.2.7 to change the maximum boron carbide content in METAMIC to 33.0 weight percent.

### **Reason and Justification for Proposed Change**

The results of METAMIC pre-production runs in the fabrication facility indicate that the target boron carbide content needed to be established at 32.0 weight percent in order to provide reasonable assurance of repeatability is achieving the minimum required  $^{10}\text{B}$  loading in the neutron absorber. The previous CoC maximum proposed value of 32.5 weight percent, if implemented, created a very

small range of acceptability and risked unnecessarily rejecting otherwise acceptable neutron absorber material. The new value is only 0.5 weight percent than the previously proposed value, which maintains it reasonably close to the 31.0 weight percent value for the METAMIC studied in the EPRI test program, used as the basis for this change.

### **Proposed Change No. 25b**

CoC Appendix B, Section 3.2.8 (new) and FSAR Section 9.1.5.3:

Add new Section 3.2.8 to incorporate FSAR Section 9.1.5.3 into the CoC by reference. Add a note in the FSAR that this section may not be modified under the provisions of 10 CFR 72.48.

#### **Reason and Justification for Proposed Change**

Second round RAI Number 12-1 requested the neutron absorber testing requirements specified in FSAR Section 9.1.5.3 to be incorporated into the CoC by reference so that the NRC can control any proposed changes to these requirements. The change to the FSAR will preclude any inadvertent changes to this information under 10 CFR 72.48, which would not apply because the information is officially part of the CoC.

### **Proposed Change No. 26**

CoC Appendix B, Section 3.4.3

Clarify the manner in which the equation used to determine whether the site may deploy free-standing casks is executed, as shown in the attached marked-up CoC.

#### **Reason for Proposed Change**

Use of Zero Period Accelerations (ZPAs) in this equation is unnecessarily conservative and an alternative approach has been requested by a HI-STORM System user. In addition, two criteria must be met with this equation, namely, incipient sliding, where the value of “ $\mu$ ” is the coefficient of friction between the overpack and the ISFSI pad, and incipient tipping, where the value of “ $\mu$ ” is the ratio of the cask radius to the height of the cask center-of-gravity above the ISFSI pad surface.

### **Justification for Proposed Change**

The intent of the equation is to verify that there will be no incipient tipping or sliding of the cask under the site-specific seismic condition at the ISFSI. Therefore, both definitions of “ $\mu$ ” must be evaluated. Use of ZPA values is a bounding approach, but may be overly conservative, particularly for those site where the ISFSI may already be constructed. See proposed changes to FSAR Section 3.4.7.1 for the detailed justification.

### **Proposed Change No. 27**

Certificate of Compliance, Appendix B, Section 3.6.3

Modify the FSAR section numbers called out in this technical specification to state Section 4.4, rather than the detailed subsection numbers.

### **Reason and Justification for Proposed Change**

The level of specificity of the existing section numbers is unnecessary.

### **Proposed Change No. 28**

FSAR Section 2.0.2 and Tables 1.0.3, 2.0.2, 1.D.1 and 2.2.3

Modify the design temperatures of the MPC shell, overpack concrete, and Holtite neutron shield material as shown in the attached markup of FSAR Appendix 1.D and Section 2.2.

### **Reason and Justification for Proposed Changes**

These design temperature increases *provide the greatest assurance that the structural evaluation in FSAR Chapter 3 bound current and potential future conditions.* ~~are necessary as a consequence of the revised thermal analysis, where calculated temperatures exceeded the previous design temperatures.~~ This change expands existing Proposed Change No. 16b.

The increase in the MPC shell normal design temperature has been evaluated and found to be acceptable from a structural perspective (see FSAR Sections 3.1 and 3.4.4.3.1.2 and Table 3.4.6 for the results of the structural evaluations of this change).

The increase in the overpack concrete temperature limits and the change to the limit applicability to “through-thickness section average” is based on Appendix A to ACI 349. Specifically, Paragraph A.4.3 of ACI 349 Appendix A allows the use of elevated temperature limits if test data supporting the compressive strength is available and an evaluation showing no concrete deterioration is provided. For short-term conditions, the through-thickness section average concrete temperature limit is specified in accordance with Paragraph A.4.2 of ACI 349, Appendix A. The required evaluations and a description of the test data are available for inspection at Holtec’s offices. This change was implemented under the provisions of 10 CFR 72.48 and is included in this LAR submittal because it is germane to the thermal analyses of the increased heat duty for the cask system.

The creation of a short-term temperature limit for Holtite-A, which is used only in the HI-TRAC transfer cask, is based on test data summarized in Holtec Report HI-2002396, Revision 3. This report was submitted to the NRC in May, 2003 on Docket 71-9261.

### **Proposed Change No. 29**

#### **FSAR Tables 2.2.6 and 2.2.7:**

The above-referenced SAR tables are proposed to be modified to clarify the Code applicability for the MPC basket and basket angle supports. The MPC basket and basket angle supports are governed by ASME III, Subsection NG. This change clarifies that, based on their design function, the basket is considered a core support structure pursuant to Article NG-1121 and the angle supports are considered internal structures pursuant to Article NG-1122.

#### **Reason for Proposed Change**

To remove ambiguity regarding the applicability of ASME Section III, Subsection NG, Article NG-1120 to these components.

#### **Justification for Proposed Change**

Article NG-1121 defines core support structures as “those structures or parts of structures which are designed to provide direct [emphasis added] support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel. Structures which support or restrain the core only after the postulated failure of core support structures are considered to be internal structures.”

Article NG-1122 defines internal structures as “all structures within the reactor pressure vessel other than core support structures.” The MPC fuel basket provides direct support of the fuel assemblies appropriately classified as a core support



structure under Article NG-1121. The MPC basket angle supports do not provide direct support of the fuel and are, therefore, classified as internal structures under Article NG-1122.

**Proposed Change No. 30**

FSAR Section 11.1 and 11.2:

- a. Modify the FSAR to add FHD System Failure and SCS Power Failure as new off-normal events 11.1.6 and 11.1.7.
- b. Modify the FSAR to add Supplemental Cooling System Failure as a new accident event.

**Reason and Justification for Proposed Change**

Based on the design functions performed by these not-important-to-safety systems, it was deemed prudent to postulate appropriate off-normal and accident events.

***SECTION IV – NEW CHANGES SUBSEQUENT TO SECOND ROUND RAI  
RESPONSES***

***Proposed Change No. 31***

*Certificate of Compliance, Appendix B, Section 3.4:*

*Insert a new requirement after Item 3a to address degraded cask/pad interface friction for freestanding casks.*

***Reason and Justification for Proposed Change***

*This change is proposed to clarify the requirements for site-specific evaluation of cask/pad interface friction for freestanding casks. It is possible for ice or moisture to collect beneath a cask, which would reduce the coefficient of friction at the cask/pad interface. Such a condition could alter the seismic response of the casks. Response of casks to such a condition is necessarily site-specific and must be evaluated on a site-specific basis. The proposed new requirements ensure that any such condition is properly and uniformly evaluated.*

**LAR 1014-2, REVISION 2**

**ATTACHMENT 2**

**COC CHANGES – MARKUP VERSION**

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**

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

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1014	05/31/00	06/01/20	72-1014	42		USA/72-1014

Issued To: (Name/Address)  
  
Holtec International  
Holtec Center  
555 Lincoln Drive West  
Marlton, NJ 08053

Safety Analysis Report Title  
  
Holtec International Inc., Final Safety Analysis Report for the HI-STORM 100 Cask System  
Docket No. 72-1014

**CONDITIONS**

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

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 400 or 400S), 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 (PWR), fuel assemblies or 68 boiling water reactor (BWR) 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. The cask comprises three discrete components: the MPCs, the HI-TRAC transfer cask, and the HI-STORM 400 or 400S 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. It is made entirely of stainless steel except for the neutron absorbers and optional aluminum heat conduction elements (AHCEs), which are installed in some early-vintage MPCs. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with Boron neutron absorbers, provides criticality control.

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

1. b. Description (continued)

There are ~~seven~~ *eight* types of MPCs: the MPC-24, MPC-24E, MPC-24EF, MPC-32, ~~MPC-32F~~, MPC-68, MPC-68F, and MPC-68FF. *The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC.* The MPC-24 and MPC-32 hold up to 24 and 32 PWR fuel assemblies, respectively, that must be intact. The MPC-24E holds up to 24 PWR fuel assemblies, up to four of which may be classified as damaged fuel assemblies. The MPC-24EF holds up to 24 PWR fuel assemblies, up to four of which may be classified as damaged fuel assemblies or in the form of fuel debris. The MPC-68 holds up to 68 BWR fuel assemblies that may be intact or damaged (i.e., with known or suspected cladding defects greater than hairline cracks or pinholes). The number of damaged fuel assemblies is limited to sixteen unless they are Dresden Unit 1 or Humboldt Bay fuel assemblies. The MPC-68F holds up to 68 Dresden Unit 1 or Humboldt Bay BWR fuel assemblies that may be intact, damaged, with up to four in the form of fuel debris (i.e., with known or suspected defects such as ruptured fuel rods, severed fuel rods, and loose fuel pellets). The MPC-68FF holds up to 68 BWR fuel assemblies, up to sixteen of which may be classified as damaged fuel or fuel debris. A maximum of eight fuel assemblies may be in the form of fuel debris. All fuel to be stored in the HI-STORM 100 System must comply with the limits specified in Appendix B to this GoC. All ~~seven~~ *eight* MPC models have the same external dimensions ~~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 water jacket attached to the exterior. Two ~~types~~ *typesizes* of HI-TRAC transfer casks are available: the 125 ton-HI-TRAC and the 100 ton HI-TRAC. The weight designation is the maximum weight of a loaded transfer cask during any loading, unloading or transfer operation. Both transfer cask ~~types~~ *typesizes* have identical cavity diameters. The 125 ton HI-TRAC transfer cask has thicker lead and water shielding and larger outer dimensions than the 100 ton HI-TRAC transfer cask.

The HI-STORM 100 or 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S is a ~~shortened version~~ *variation* of the *HI-STORM 100 overpack design that includes with a modified lid design incorporating 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 a *minimum* of four air inlets at the bottom and a *minimum* of four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has channels attached to its interior surface to guide the MPC during insertion and removal, provide a flexible 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 is a variant of the HI-STORM 100 family and is outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the concrete storage pad in high seismic applications. The HI-STORM 100A applies to both the ~~standard~~ (HI-STORM 100) and ~~short~~ (HI-STORM 100S) overpacks that are classified as the HI-STORM 100A and HI-STORM 100SA, respectively.

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.

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

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 a HI-STORM 100 or 100S 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 ~~safety~~ *regulatory* 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/or Sections 3.4.6 and Section 3.5 of Appendix B to this certificate, as applicable.

6. APPROVED CONTENTS

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B 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 Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM Cask Systems (for each thermally unique MPC basket design - MPC-24/24E/24EF, MPC-32/32F, and MPC-68/68F/68FF) placed into service by any user with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR.

Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.

Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

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.
- 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.
- f. MPC welding, NDE inspections, ~~hydrostatic pressure~~ testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), ~~and helium backfilling, and leakage testing.~~ (A mockup may be used for this dry-run exercise.)
- g. Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.
- h. Transfer of the MPC from the transfer cask to the overpack.
- i. Placement of the HI-STORM 100 Cask System at the ISFSI.
- j. HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

11. 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, and the attached Appendix A and Appendix B. ~~At~~*The HI-STORM 100 Cask Systems must 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 different CoC amendments, may be used with one another provided the CoC does not specifically prohibit their use with each other, and a design compatibility assessment is performed by the CoC holder. Amendment No. 1; except that general licensees may use the HI-STORM 100 Cask Systems that were fabricated in accordance with the original CoC.*

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director  
Spent Fuel Project Office  
Office of Nuclear Material Safety  
and Safeguards

Attachments:

1. Appendix A
2. Appendix B





**CERTIFICATE OF COMPLIANCE NO. 1014**

**APPENDIX A**

**TECHNICAL SPECIFICATIONS  
FOR THE HI-STORM 100 CASK SYSTEM**

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## 1.0 USE AND APPLICATION

### 1.1 Definitions

---

-----NOTE-----

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

-----

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
FUEL BUILDING	The FUEL BUILDING is the site-specific power plant facility, governed by the regulations of 10CFR Part 50, where the loaded OVERPACK or TRANSFER CASK is transferred to or from the transporter.
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 included MPC transfer between the TRANSFER CASK and the OVERPACK.
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.

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(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 OVERPACK does not include the TRANSFER CASK.
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 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 sitting on a storage pad within the ISFSI perimeter. STORAGE OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
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. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.

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(continued)

## 1.1 Definitions (continued)

---

### TRANSPORT OPERATIONS

TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS includes transfer of the MPC between the OVERPACK and the TRANSFER CASK.

### UNLOADING OPERATIONS

UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.

---

## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

---

**PURPOSE**            The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

---

**BACKGROUND**        Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

---

(continued)

1.2 Logical Connectors

---

EXAMPLES      The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . .  <u>AND</u>  A.2 Restore . . .	

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

(continued)

---

1.2 Logical Connectors

EXAMPLES  
(continued)

EXAMPLE 1.2-2

ACTIONS

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

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

OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.



1.0 USE AND APPLICATION

1.3 Completion Times

---

**PURPOSE**            The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

---

**BACKGROUND**      Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

---

**DESCRIPTION**      The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the HI-STORM 100 System is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the HI-STORM 100 System is not within the LCO Applicability.

Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

---

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-3

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times (continued)

---

IMMEDIATE  
COMPLETION  
TIME

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

---

## 1.0 USE AND APPLICATION

### 1.4 Frequency

---

**PURPOSE**            The purpose of this section is to define the proper use and application of Frequency requirements.

---

**DESCRIPTION**      Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

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(continued)

1.4 Frequency (continued)

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EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

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

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

(continued)

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1.4 Frequency

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EXAMPLES  
(continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

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

---



2.0

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This section is intentionally left blank

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**3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY**

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LCO 3.0.1            LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.

---

LCO 3.0.2            Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

---

LCO 3.0.3            Not applicable.

---

LCO 3.0.4            When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.

---

LCO 3.0.5            Equipment removed from service or not in service in compliance with ACTIONS maybe returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

---

### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

SR 3.0.1            SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.

---

SR 3.0.2            The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply. If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

---

SR 3.0.3            If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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#### SR 3.0.3

(continued)

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

---

#### SR 3.0.4

Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

---

3.1 SFSC INTEGRITY

3.1.1 Multi-Purpose Canister (MPC)

LCO 3.1.1 The MPC shall be dry and helium filled.

APPLICABILITY: During TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure or demohsturizer exit gas temperature limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u> A.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	30 days
B. MPC helium backfill limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential.	72 hours
	<u>AND</u> B.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	14 days



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	<p>For those MPCs containing all moderate burnup (<math>\leq 45,000</math> MWD/MTU) fuel assemblies, verify MPC cavity vacuum drying pressure is within the limit specified in Table 3-1 for the applicable MPC model.</p> <p><u>OR</u></p> <p>For those MPCs containing fuel assemblies of any authorized burnup, while using the recirculating helium method to dehydrate the MPC cavity, verify that the gas temperature exiting the demister is <math>\leq 21^{\circ}\text{F}</math> for <math>\geq 30</math> minutes.</p> <p><i>Verify that the MPC cavity has been dried in accordance with the applicable limits in Table 3-1.</i></p>	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill density or pressure quantity is within the limit specified in Table 3-42 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.3	Verify that the total helium leak rate through the MPC lid confinement weld and the drain and vent port confinement welds is less than the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be operable

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System inoperable.	A.1 Restore SFSC Heat Removal System to operable status.	8 hours
B. Required Action A.1 and associated Completion Time not met.	B.1 <del>Perform SR 3.2.3.1. Measure SFSC dose rates in accordance with the Radiation Protection Program.</del>	Immediately and once per 12 hours thereafter
	<u>AND</u>	
	B.2.1 Restore SFSC Heat Removal System to operable status.	4864 hours
	<u>OR</u>	
	B.2.2 Transfer the MPC into a TRANSFER CASK.	4864 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify all OVERPACK inlet and outlet air ducts are free of blockage.	24 hours
	<u>OR</u>	
	For OVERPACKS with installed temperature monitoring equipment, verify that the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is $\leq 126^{\circ}\text{F}$ .	24 hours

3.1 SFSC INTEGRITY

3.1.3 Fuel Cool-Down

LCO 3.1.3 The MPC *cavity bulk* helium exit temperature shall be  $\leq 200^\circ$  F

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

APPLICABILITY: UNLOADING OPERATIONS prior to re-flooding.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC <i>cavity bulk</i> helium <del>gas exit</del> temperature not within limit.	A.1 Establish MPC <i>cavity bulk</i> helium <del>gas exit</del> temperature within limit.  <u>AND</u> A.2 Ensure adequate heat transfer from the MPC to the environment	Prior to initiating MPC re-flooding operations  <del>22 hours</del> <i>Immediately</i>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 <del>Verify</del> <i>Ensure via analysis or direct measurement of MPC exit gas temperature that MPC cavity bulk helium <del>gas exit</del> temperature is within limit.</i>	Prior to MPC re-flooding operations.

3.1 SFSC INTEGRITY

3.1.4 Supplemental Cooling System

LCO 3.1.4 The Supplemental Cooling System (SCS) shall be operable

-----NOTE-----

Upon reaching steady state operation, the SCS may be temporarily disabled for a short duration ( $\leq 7$  hours) to facilitate necessary operational evolutions, such as movement of the TRANSFER CASK through a door way, or other similar operation.

-----

APPLICABILITY: This LCO is applicable when the loaded MPC is in the TRANSFER CASK and:

a. Within 4 hours of the completion of MPC drying operations in accordance with LCO 3.1.1 or within 4 hours of transferring the MPC into the TRANSFER CASK if the MPC is to be unloaded

AND

b. The MPC contains one or more fuel assemblies with an average burnup  $> 45,000$  MWD/MTU

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Supplemental Cooling System inoperable.	A.1 Restore SFSC Supplemental Cooling System to operable status.	7 days
B. Required Action A.1 and associated Completion Time not met.	B.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify Supplemental Cooling System is operable.	2 hours

3.2 SFSC RADIATION PROTECTION ~~Deleted.~~

3.2.1 TRANSFER CASK Average Surface Dose Rates ~~Deleted.~~

LCO 3.2.1 ~~Deleted.~~ The average surface dose rates of each TRANSFER CASK shall not exceed:

- ~~\_\_\_\_\_ a. 125 Ton TRANSFER CASK~~
  - ~~\_\_\_\_\_ i. 220 mrem/hour (neutron + gamma) on the side;~~
  - ~~\_\_\_\_\_ ii. 60 mrem/hour (neutron + gamma) on the top~~
- ~~\_\_\_\_\_ b. 100 Ton TRANSFER CASK~~
  - ~~\_\_\_\_\_ i. 1500 mrem/hour (neutron + gamma) on the side;~~
  - ~~\_\_\_\_\_ ii. 315 mrem/hour (neutron + gamma) on the top~~

APPLICABILITY: ~~During TRANSPORT OPERATIONS.~~

ACTIONS

~~----- NOTE -----~~

~~-----~~

~~Separate Condition entry is allowed for each TRANSFER CASK.~~

~~-----~~

~~-----~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
<del>A. TRANSFER CASK average surface dose rate limits not met.</del>	<del>A.1 Administratively verify correct fuel loading.</del>	<del>24 hours</del>
	<del><u>AND</u></del> <del>A.2 Perform a written evaluation to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.</del>	<del>48 hours</del>

~~(continued)~~

~~Deleted TRANSFER CASK Average Surface Dose Rates~~  
3.2.1

~~ACTIONS~~  
~~(continued)~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
<del>B. Required Action and associated Completion Time not met.</del>	<del>B.1 Remove all fuel assemblies from the TRANSFER CASK</del>	<del>30 days</del>

~~SURVEILLANCE REQUIREMENTS~~

SURVEILLANCE	FREQUENCY
<p><del>SR 3.2.1.1</del> — Verify average surface dose rates of the TRANSFER CASK loaded with an MPC containing fuel assemblies are within limits.</p> <p>————— A minimum of 12 dose rate measurements shall be taken on the side of the TRANSFER CASK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively. Within each set, the measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket. The average of the 12 dose rate measurements shall be compared to the limit specified in LCO 3.2.1.a.i or b.i, as applicable.</p> <p>————— A minimum of four (4) top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid. The average of these four dose rates shall be compared to the limit specified in LCO 3.2.1.a.ii or b.ii, as applicable.</p>	<p><del>Once, prior to TRANSPORT OPERATIONS</del></p>

Transfer Cask Average Surface Dose Rates  
3.2.1

————— ~~FIGURE 3.2.1-1 INTENTIONALLY DELETED~~

3.2 SFSC RADIATION PROTECTION

3.2.2 TRANSFER CASK Surface Contamination

LCO 3.2.2           Removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC shall each not exceed:

- a. 1000 dpm/100 cm<sup>2</sup> from beta and gamma sources
- b. 20 dpm/100 cm<sup>2</sup> from alpha sources.

-----NOTE-----

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

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APPLICABILITY:    During TRANSPORT OPERATIONS.

ACTIONS

-----NOTE-----

-  
Separate Condition entry is allowed for each TRANSFER CASK.

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify that the removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC containing fuel is within limits.	Once, prior to TRANSPORT OPERATIONS



3.2 SFSC RADIATION PROTECTION ~~Deleted.~~

3.2.3 OVERPACK Average Surface Dose Rates ~~Deleted.~~

LCO 3.2.3 ~~Deleted. The average surface dose rates of each OVERPACK shall not exceed:~~

- ~~\_\_\_\_\_ a. \_\_\_\_\_ 50 mrem/hour (neutron + gamma) on the side~~
- ~~\_\_\_\_\_ b. \_\_\_\_\_ 10 mrem/hour (neutron + gamma) on the top~~
- ~~\_\_\_\_\_ c. \_\_\_\_\_ 45 mrem/hour (neutron + gamma) at the inlet and outlet vent ducts~~

~~APPLICABILITY: \_\_\_\_\_ During STORAGE OPERATIONS.~~

~~ACTIONS~~

~~NOTE~~

~~Separate Condition entry is allowed for each SFSC.~~

<del>CONDITION</del>	<del>REQUIRED ACTION</del>	<del>COMPLETION TIME</del>
<del>A. OVERPACK average surface dose rate limits not met.</del>	<del>A.1 Administratively verify correct fuel loading.</del>  <del>AND</del>  <del>A.2 Perform a written evaluation to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.</del>	<del>24 hours</del>  <del>48 hours</del>
<del>B. Required Action and associated Completion Time not met.</del>	<del>B.1 Remove all fuel assemblies from the SFSC.</del>	<del>30 days</del>

~~Deleted OVERPACK Average Surface Dose Rates~~  
3.2.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.3.1 — Verify average surface dose rates of the OVERPACK loaded with an MPC containing fuel assemblies are within limits.</p> <p>————— A minimum of 12 dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask. The average of the 12 dose rate measurements shall be compared to the limit specified in LCO 3.2.3.a.</p> <p>————— A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top shield, 90 degrees apart around the circumference of the lid. The average of the 5 dose rate measurements shall be compared to the limit specified in LCO 3.2.3.b.</p> <p>————— A dose rate measurement shall be taken adjacent to each inlet and outlet vent duct. The average of the 8 inlet and outlet duct dose rates shall be compared to the limit specified in LCO 3.2.3.c.</p>	<p>Once, within 24 hours after beginning STORAGE OPERATIONS</p> <hr/>

OVERPACK Average Surface Dose Rates  
3.2.3

Figure 3.2.3-1 INTENTIONALLY DELETED

3.3 SFSC CRITICALITY CONTROL

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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%  $^{235}\text{U}$ :  $\geq 400$  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%  $^{235}\text{U}$ :  $\geq 300$  ppmb
  - c. ~~MPC-32 with all fuel assemblies having an initial enrichment  $\leq 4.1$  wt%  $^{235}\text{U}$ :  $\geq 1900$  ppmb~~
  - d. ~~MPC-32 with one or more fuel assemblies having an initial enrichment  $> 4.1$  and  $\leq 5.0$  wt%  $^{235}\text{U}$ :  $\geq 2600$  ppmb~~
  - 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%  $^{235}\text{U}$  and  $\leq 5.0$  wt%  $^{235}\text{U}$ :  $\geq 600$  ppmb
  - f. MPC-32/32F: Minimum soluble boron as required by the table below.

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

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

AND

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

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- This surveillance is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. -----	Once, within 4 hours of <i>prior to</i> entering the Applicability of this LCO.
SR 3.3.1.1 Verify boron concentration is within the applicable limit using two independent measurements.	<u>AND</u> Once per 48 hours thereafter.

Table 3-1  
MPC Cavity Drying Limits

<b>Fuel Burnup (MWD/MTU)</b>	<b>MPC Heat Load (kW)</b>	<b>Method of Moisture Removal (Notes 1 and 2)</b>
All Assemblies $\leq 45,000$	$\leq 28.74$	VDS or FHD
One or more assemblies $> 45,000$	$\leq 28.74$	FHD

Notes:

1. VDS means Vacuum Drying System. The acceptance criterion for VDS is MPC cavity pressure shall be  $\leq 3$  torr for  $\geq 30$  minutes.
2. FHD means Forced Helium Dehydration System. The acceptance criterion for the FHD System is gas temperature exiting the demoinsturizer shall be  $\leq 21^{\circ}\text{F}$  for  $\geq 30$  minutes or gas dew point exiting the MPC shall be  $\leq 22.9^{\circ}\text{F}$  for  $\geq 30$  minutes .

Table 3-42

MPC Helium Backfill Model-Dependent Limits<sup>1</sup>

MPC MODEL	LIMITS
<del>1. MPC-24/24E/24EF</del>	
<del>a. MPC Cavity Vacuum Drying Pressure</del>	<del><math>\leq 3</math> torr for <math>\geq 30</math> min</del>
<del>b. MPC Helium Backfill<sup>†</sup></del>	<del></del>
<del><i>Cask Heat Load <math>\leq 27.77</math> kW (MPC-24) or <math>\leq 28.17</math> kW (MPC-24E/EF)</i></del>	<del>0.1212 +0/-10% g-moles/l OR <math>\geq 29.3</math> psig and <math>\leq 33.3</math> psig</del>
<del>c. MPC Helium Leak Rate</del>	<del><math>\leq 5.0E-6</math> atm cc/sec (He)</del>
<del>2. MPC-68/68F/68FF</del>	
<del>a. MPC Cavity Vacuum Drying Pressure</del>	<del><math>\leq 3</math> torr for <math>\geq 30</math> min</del>
<del>b. MPC Helium Backfill<sup>†</sup></del>	<del></del>
<del><i>Cask Heat Load <math>\leq 28.19</math> kW</i></del>	<del>0.1218 +0/-10% g-moles/l OR <math>\geq 29.3</math> psig and <math>\leq 33.3</math> psig</del>
<del>c. MPC Helium Leak Rate</del>	<del><math>\leq 5.0E-6</math> atm cc/sec (He)</del>
<del>3. MPC-32/32F</del>	
<del>a. MPC Cavity Vacuum Drying Pressure</del>	<del><math>\leq 3</math> torr for <math>\geq 30</math> min</del>
<del>b. MPC Helium Backfill Pressure<sup>1</sup></del>	<del></del>
<del><i>Cask Heat Load <math>\leq 28.74</math> kW</i></del>	<del><math>\geq 29.3</math> psig and <math>\leq 33.3</math> psig</del>
<del>c. MPC Helium Leak Rate</del>	<del><math>\leq 5.0E-6</math> atm cc/sec (He)</del>

<sup>1</sup> Helium used for backfill of MPC shall have a purity of  $\geq 99.995\%$ . Pressure range is at a reference temperature of 70°F

4.0

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## 5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

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The following programs shall be established, implemented and maintained.

5.1 Deleted.

5.2 Deleted.

5.3 Deleted.

5.4 Radioactive Effluent Control Program

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

- a. The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.
- b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.
- c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

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(continued)

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 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.).

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, 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. For site-specific transport route surfaces that are not bounded by either the Set A or Set B parameters of FSAR Table 2.2.9, 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)

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## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 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 transportation between the FUEL BUILDING and the CTF and/or ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 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 transportation between the FUEL BUILDING and the CTF and/or ISFSI pad provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.

(continued)

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ADMINISTRATIVE CONTROLS AND PROGRAMS

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5.5 Cask Transport Evaluation Program (continued)

Table 5-1

TRANSFER CASK and Free-Standing OVERPACK Lifting Requirements

ITEM	ORIENTATION	LIFTING HEIGHT LIMIT (in.)
TRANSFER CASK	Horizontal	42 (Notes 1 and 2)
TRANSFER CASK	Vertical	None Established (Note 2)
OVERPACK	Horizontal	Not Permitted
OVERPACK	Vertical	11 (Note 3)

- Notes:
1. To be measured from the lowest point on the TRANSFER CASK (i.e., the bottom edge of the cask/lid assemblage)
  2. See Technical Specification 5.5.a.3 and 4
  3. See Technical Specification 5.5.a.3.

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(continued)

ADMINISTRATIVE CONTROLS AND PROGRAMS

5.6 ~~Fuel Cladding Oxide Thickness Evaluation Program Deleted.~~

~~A Fuel Cladding Oxide Thickness Evaluation Program shall be developed and implemented to determine the average fuel cladding oxide thickness of high burnup (> 45,000 MWD/MTU) spent nuclear fuel assemblies proposed to be stored in the HI-STORM 100 System. The program may use direct physical measurements or an appropriate predictive methodology with due consideration of all significant variables (e.g., in-core flux, cycle length and number, power history, coolant temperature profile, coolant chemistry, and metallurgy of the fuel cladding material) to determine the average oxide thickness on the fuel cladding. If a predictive methodology is used to determine average fuel cladding oxide thickness, a sufficient number of fuel cladding thickness measurements shall be made to adequately benchmark the methodology.~~

~~In order to classify a high burnup spent fuel assembly as an INTACT FUEL ASSEMBLY, the loss of fuel cladding to oxidation must not increase the fuel cladding inner radius-to-fuel cladding thickness ratio above 10.5 for PWR fuel assemblies or 9.5 for BWR fuel assemblies. The criterion is met if the computed or measured average oxidation layer thickness of all fuel rods is less than the maximum allowable average fuel cladding oxidation thickness. The maximum allowable average fuel cladding oxidation layer thickness shall be calculated using the following formula:~~

$$t_{ox} = \left( t_{nom} - \frac{0.5 \times d_{nom} - t_{nom}}{W} \right) \times 25,400$$

~~where:~~

~~$t_{ox}$  = the maximum allowable average oxidation layer thickness (micrometers)~~

~~$W$  = the applicable maximum allowable fuel cladding inner radius-to-fuel cladding thickness ratio (10.5 or 9.5)~~

~~$t_{nom}$  = the nominal, pre-irradiated fuel cladding thickness (inches)~~

~~$d_{nom}$  = the nominal, pre-irradiated fuel cladding outer diameter (inches)~~

~~A high burnup spent fuel assembly shall be considered a DAMAGED FUEL ASSEMBLY if the computed or measured average oxidation layer thickness on any fuel rod exceeds the limit determined above.~~

(continued)

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 5.7 Radiation Protection Program

- 5.7.1 *Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for measuring dose rates, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). This program provides a means to help ensure that licensees using the HI-STORM 100 System do not violate the dose limits in 10 CFR 72. The actions and criteria to be included in the program are provided below.*
- 5.7.2 *As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.*
- 5.7.3 *Based on the analysis performed pursuant to Section 5.7.2, the licensee shall establish cask surface dose rate limits for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be used at the site. Average neutron and gamma dose rate limits shall be established at the following locations:*
- a. *The top of the TRANSFER CASK and the OVERPACK.*
  - b. *The side of the TRANSFER CASK and OVERPACK*
  - c. *The average of the inlet and outlet ducts on the OVERPACK*
- 5.7.4 *Notwithstanding the limits established in Section 5.7.3, the measured dose rates on a loaded OVERPACK shall not exceed the following values:*
- a. *30 mrem/hr (gamma + neutron) on the top of the OVERPACK*
  - b. *125 mrem/hr (gamma + neutron) on the side of the OVERPACK*
  - c. *130 mrem/hr (gamma + neutron) at the inlet and outlet vent ducts*
- 5.7.5 *The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates and calculate average values as described in Section 5.7.8 and 5.7.9 for comparison against the limits established in Section 5.7.3 or Section 5.7.4, whichever are lower.*

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 5.7 Radiation Protection Program (cont'd)

- 5.7.6 *If the measured average surface dose rates exceed the lower of the two limits established in Section 5.7.3 or Section 5.7.4, the licensee shall:*
- a. *Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.*
  - b. *Perform an evaluation to verify whether placement of the as-loaded OVERPACK at the ISFSI will cause the dose limits of 10 CFR 72.104 to be exceeded.*
- 5.7.7 *If the evaluation performed pursuant to Section 5.7.6 shows that the dose limits of 10 CFR 72.104 will be exceeded, appropriate corrective action shall be taken to ensure the dose limits are not exceeded.*
- 5.7.8 *TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:*
- a. *A minimum of 12 dose rate measurements shall be taken on the side of the TRANSFER CASK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively. Within each set, the measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.*
  - b. *A minimum of four (4) TRANSFER CASK top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid.*
  - c. *A minimum of 12 dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.*
  - d. *A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top shield, 90 degrees apart around the circumference of the lid.*

ADMINISTRATIVE CONTROLS AND PROGRAMS

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5.7 Radiation Protection Program (cont'd)

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

5.7.9 *The average measured dose rates shall be calculated by summing the individual neutron and gamma dose rates measured in Sections 5.7.8.a through 5.7.8.e and dividing by the total number of measurements for that section. The neutron and gamma dose rates shall be averaged separately.*

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**CERTIFICATE OF COMPLIANCE NO. 1014**

**APPENDIX B**

**APPROVED CONTENTS AND DESIGN FEATURES**

**FOR THE HI-STORM 100 CASK SYSTEM**

## 1.0 Definitions

## -----NOTE-----

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

<u>Term</u>	<u>Definition</u>
CASK TRANSFER FACILITY (CTF)	The CASK TRANSFER FACILITY includes the following components and equipment: (1) a Cask Transfer Structure used to stabilize the TRANSFER CASK and MPC during lifts involving spent fuel not bounded by the regulations of 10 CFR Part 50, and (2) Either a stationary lifting device or a mobile lifting device used in concert with the stationary structure to lift the OVERPACK, TRANSFER CASK, and MPC
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STORM 100 System are as follows: <ol style="list-style-type: none"> <li>1. Holtec Dresden Unit 1/Humboldt Bay design</li> <li>2. Transnuclear Dresden Unit 1 design</li> <li>3. Holtec Generic BWR design</li> <li>4. Holtec Generic PWR design</li> </ol>
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

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(continued)

1.0 Definitions (continued)

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INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).
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 included MPC transfer between the TRANSFER CASK and the OVERPACK.
<i>MINIMUM ENRICHMENT</i>	<i>MINIMUM ENRICHMENT is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.</i>
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.
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), water displacement guide tube plugs, and, orifice rod assemblies, and vibration suppressor inserts.
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 OVERPACK does not include the TRANSFER CASK.

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(continued)

1.0 Definitions (continued)

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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.
SPENT FUEL STORAGE CASKS (SFSCs)	An SFSC is a container approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK and its integral MPC.
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. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS include transfer of the MPC between the OVERPACK and the TRANSFER CASK.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
<i>ZR</i>	<i>ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor.</i>

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2.0 APPROVED CONTENTS

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2.1 Fuel Specifications and Loading Conditions

2.1.1 Fuel To Be Stored In The HI-STORM 100 SFSC System

- a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM 100 SFSC System.
- b. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the decay heat generation limit for the stainless steel clad fuel assemblies.
- c. For MPCs partially loaded with DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, all remaining Zircaloy (or other alloy of zirconium) ZR clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the DAMAGED FUEL ASSEMBLIES. This requirement applies only to uniform fuel loading.
- d. For MPCs partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining Zircaloy (or other alloy of zirconium) ZR clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the 6x6A, 6x6B, 6x6C, 7x7A and 8x8A fuel assemblies.
- e. All BWR fuel assemblies may be stored with or without Zircaloy (or other alloy of zirconium) ZR channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without Zircaloy ZR or stainless steel channels.

2.1.2 Uniform Fuel Loading

*Any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions related to DAMAGED FUEL, FUEL DEBRIS, and NON-FUEL HARDWARE specified in the CoC. Preferential fuel loading shall be used during uniform loading (i.e., any authorized fuel assembly in any fuel storage location) whenever fuel assemblies with significantly different post-irradiation cooling times ( $\geq 1$  year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket. Regionalized fuel loading as described in Technical Specification 2.1.3 below meets the intent of preferential fuel loading.*

(continued)

## 2.0 Approved Contents

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### 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 those fuel assemblies with Zircaloy (or other alloy of zirconium) 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, and MPC-68FF models, respectively<sup>1</sup>. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in *Section 2.4.2. Tables 2.1-6 and 2.1-7*. 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 *Deviations from Cask Contents Requirements*

*Proposed alternatives to the contents listed in Section 2.0 may be authorized on a case-specific basis by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative contents should demonstrate that:*

2.3.1 *The proposed alternative contents would provide an equivalent level of safety, and*

2.3.2 *The proposed alternative contents are consistent with the applicable requirements.*

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<sup>1</sup> 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.

2.0 APPROVED CONTENTS

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2.3 Deviations from Cask Contents Requirements (cont'd)

2.3.3 Requests for case-specific NRC approval of alternatives to contents shall be submitted in accordance with 10 CFR 72.4 by the certificate holder. Case-specific alternatives approved pursuant to this section shall be incorporated permanently into the CoC by the certificate holder in accordance with 10 CFR 72.244. Requests made pursuant to this section must meet all of the following requirements:

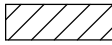
2.3.3.1 The proposed change must not significantly decrease any safety margins as described in the HI-STORM 100 System FSAR, as updated.


2.3.3.2 The proposed change may involve only the physical fuel assembly parameters listed below as specified in Tables 2.1-1, 2.1-2, and/or 2.1-3 of this Appendix:

- a. Fuel Assembly Length
- b. Fuel Assembly Width
- c. Fuel Assembly Weight
- d. Fuel Rod Clad Outside Diameter (OD)
- e. Fuel Rod Clad Inside Diameter (ID)
- f. Fuel Pellet Diameter
- g. Fuel Rod Pitch
- h. PWR Guide/Instrument Tube Thickness
- i. BWR Water Rod Thickness
- j. BWR Channel Thickness

2.3.3.3 The proposed change must be required to meet a compelling user need whereby using the normal certificate amendment process is not practical.

LEGEND:

REGION 1: 

REGION 2: 

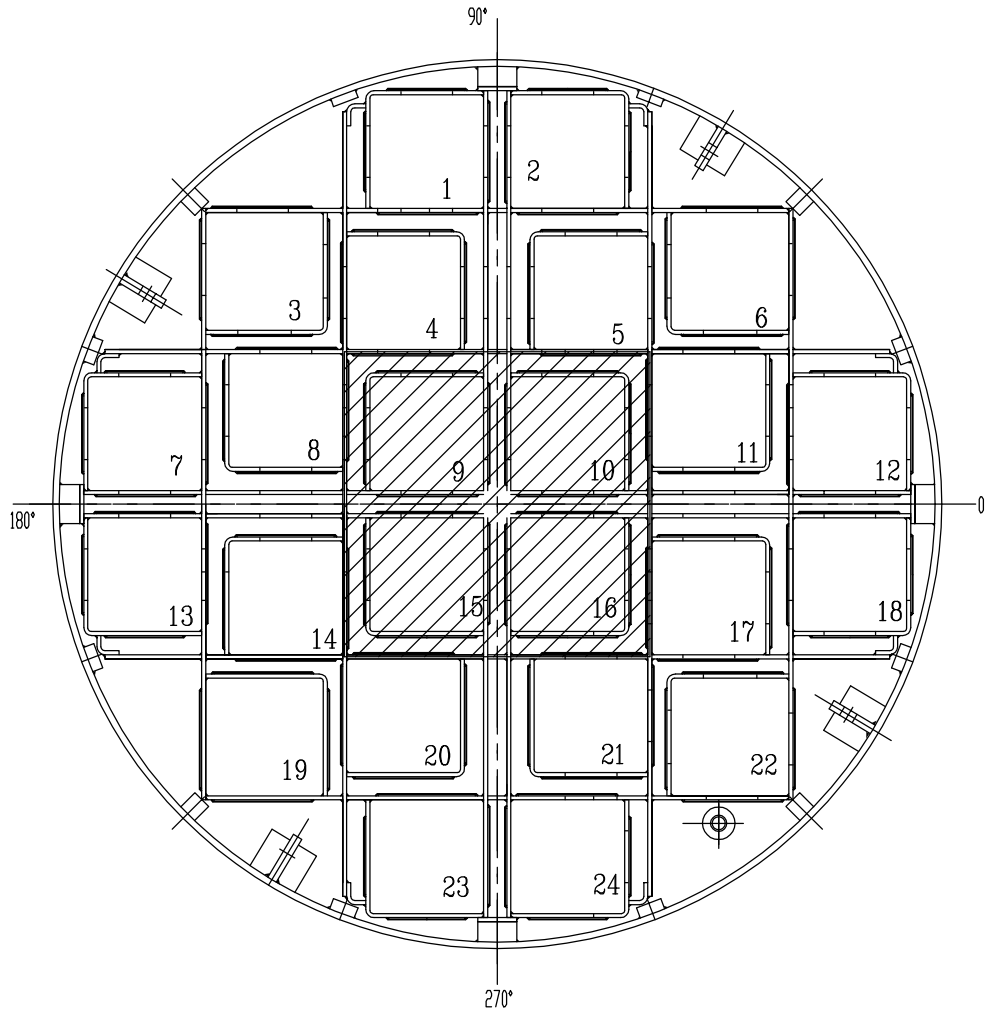
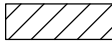



FIGURE 2.1-1  
FUEL LOADING REGIONS - MPC-24



LEGEND:

REGION 1: 

REGION 2: 

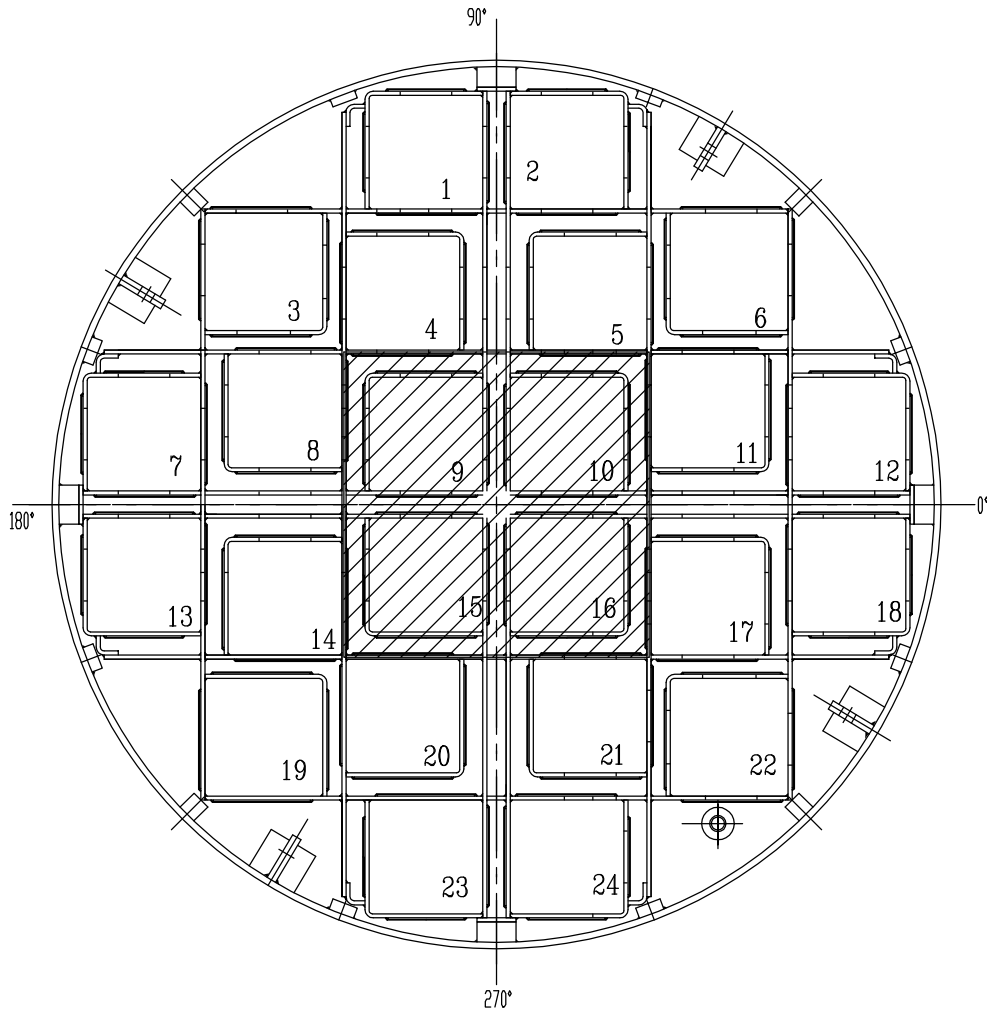

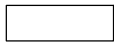


FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: 

REGION 2: 

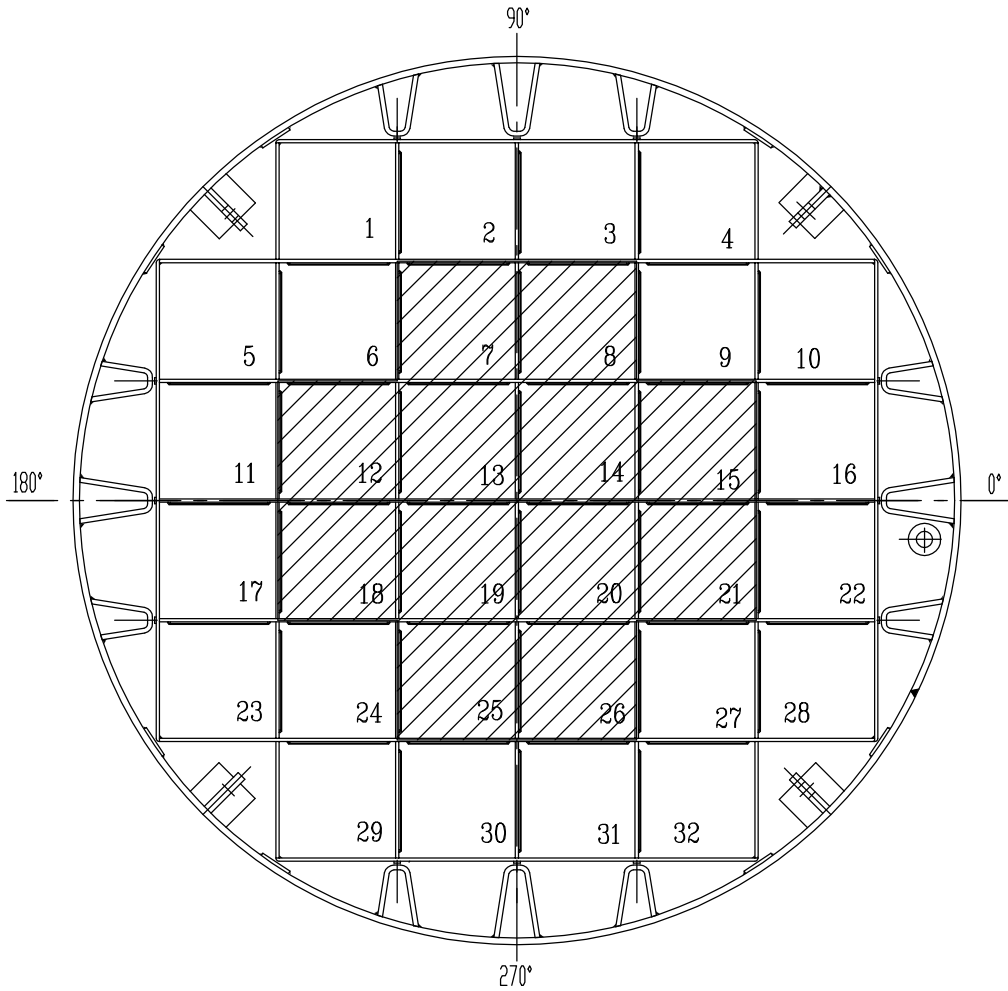




FIGURE 2.1-3

FUEL LOADING REGIONS – MPC-32/32F

LEGEND:

REGION 1: 

REGION 2: 

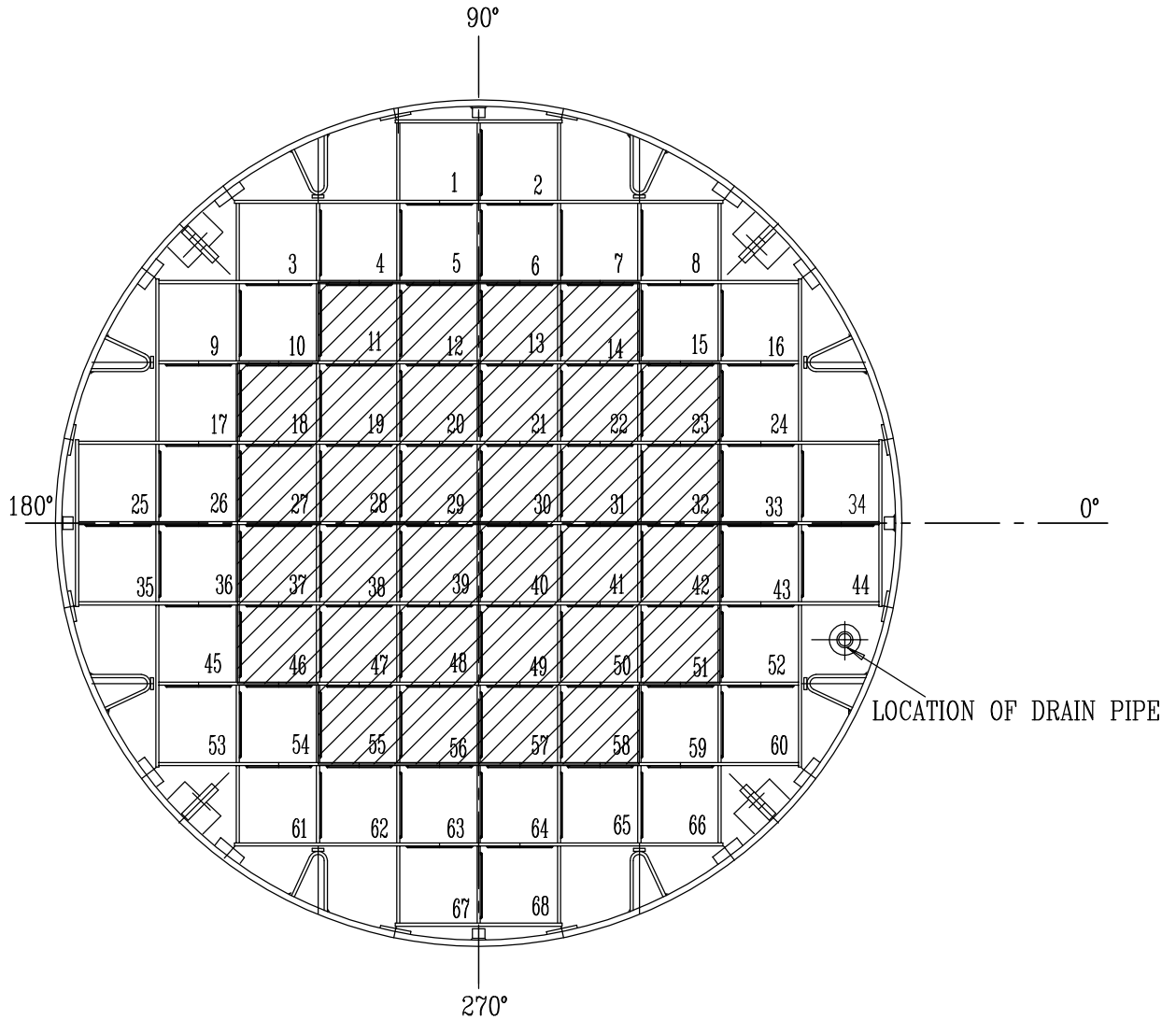


FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of 339)  
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:	Zircaloy (Zr) 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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
--	--

ii. All Other Array/Classes	Cooling time and average burnup as specified in <i>Section 2.4. Tables 2.1-4 or 2.1-6.</i>
-----------------------------	--

iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.
------------------------	------------------------------

Table 2.1-1 (page 2 of 339)  
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per ~~Assembly~~*Fuel Storage Location*:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 710$  Watts

ii All Other Array/Classes As specified in *Section 2.4. Tables 2.1-5 or 2.1-7*

e. Fuel Assembly Length:  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width:  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight:  $\leq 1,680$  lbs (including NON-FUEL HARDWARE)

B. Quantity per MPC: Up to 24 fuel assemblies.

C. Deleted.

D. *Neutron sources*, DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading into the MPC-24.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, or orifice rod assemblies, or *vibration suppressor inserts* may be stored in any fuel ~~cell~~*storage* location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 3 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

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

- |   |  |
|---|--|
| a. Cladding Type:   | Zircaloy (Zr) 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, 6x6C, 7x7A, and 8x8A:                      | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU   |
| ii. Array/Class 8x8F  | Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.  |
| iii. Array/Classes 10x10D and 10x10E                              | Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.  |
| iv. All Other Array/Classes                                       | As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>   |

Table 2.1-1 (page 4 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

i. Array/Classes 6x6A, 6x6C,  
7x7A, and 8x8A  $\leq 115$  Watts

ii. Array/Class 8x8F  $\leq 183.5$  Watts.

iii. Array/Classes 10x10D and  
10x10E  $\leq 95$  Watts

iv. All Other Array/Classes As specified in *Section 2.4. Tables 2.1-5 or 2.1-7.*

f. Fuel Assembly Length:  $\leq 176.5$  inches (nominal design)

g. Fuel Assembly Width:  $\leq 5.85$  inches (nominal design)

h. Fuel Assembly Weight:  $\leq 700$  lbs, including channels

Table 2.1-1 (page 5 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

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

a. Cladding Type:	Zircaloy (Zr) 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:	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array/Classes specified in Table 2.1-3	4.0 wt% <sup>235</sup> 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/Classes 6x6A, 6x6C, 7x7A, and 8x8A	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array Classes	As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>



Table 2.1-1 (page 6 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

- |      |   |   |
|------|---|---|
| i.   | Array/Class 6x6A, 6x6C, 7x7A,<br>and 8x8A | $\leq$ 115 Watts  |
| ii.  | Array/Class 8x8F                          | $\leq$ 183.5 Watts  |
| iii. | Array/Classes 10x10D and<br>10x10E        | $\leq$ 95 Watts   |
| iv.  | All Other Array/Classes                   | As specified in <i>Section 2.4. Tables 2.1-5 or 2.1-7</i> |

f. Fuel Assembly Length:

- |     |  |                                      |
|-----|--|--------------------------------------|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq$ 135.0 inches (nominal design) |
| ii. | All Other Array/Classes                  | $\leq$ 176.5 inches (nominal design) |

g. Fuel Assembly Width:

- |     |  |                                     |
|-----|--|-------------------------------------|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq$ 4.70 inches (nominal design) |
| ii. | All Other Array/Classes                  | $\leq$ 5.85 inches (nominal design) |

h. Fuel Assembly Weight:

- |     |  |  |
|-----|--|--|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq$ 550 lbs, including channels and DFC |
| ii. | All Other Array/Classes                  | $\leq$ 700 lbs, including channels and DFC |

Table 2.1-1 (page 7 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | Zircaloy (Zr) ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 8 of 339)  
Fuel Assembly Limits

---

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | Zircaloy (Zr) ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for array/class 6x6B.                           |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for array/class 6x6B.                           |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                  |

Table 2.1-1 (page 9 of 339)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods ( $\text{ThO}_2$  and  $\text{UO}_2$ ) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | Zircaloy (Zr) ZR   |
| b. Composition:   | 98.2 wt.% $\text{ThO}_2$ , 1.8 wt. % $\text{UO}_2$ with an enrichment of 93.5 wt. % $^{235}\text{U}$ . |
| c. Number of Rods Per Thoria Rod Canister:  | $\leq 18$  |
| d. Decay Heat Per Thoria Rod Canister:  | $\leq 115$ Watts   |
| e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister: | A fuel post-irradiation cooling time $\geq 18$ years and an average burnup $\leq 16,000$ MWD/MTIHM.    |
| f. Initial Heavy Metal Weight:  | $\leq 27$ kg/canister  |
| g. Fuel Cladding O.D.:  | $\geq 0.412$ inches  |
| h. Fuel Cladding I.D.:  | $\leq 0.362$ inches  |
| i. Fuel Pellet O.D.:  | $\leq 0.358$ inches  |
| j. Active Fuel Length:  | $\leq 111$ inches  |
| k. Canister Weight:   | $\leq 550$ lbs, including fuel   |

Table 2.1-1 (page 10 of 339)  
Fuel Assembly Limits

---

II. MPC MODEL: MPC-68 (continued)

B. Quantity per MPC:

1. Up to one (1) Dresden Unit 1 Thoria Rod Canister;
2. Up to 68 array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS;
3. Up to sixteen (16) other BWR DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68; and/or
4. Any number of BWR INTACT FUEL ASSEMBLIES up to a total of 68.

C. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

E. FUEL DEBRIS is not authorized for loading in the MPC-68.

Table 2.1-1 (page 11 of 339)  
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without ~~Zircaloy~~ (Zr) ZR channels. Uranium oxide BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array class 6x6A, 6x6C, 7x7A or 8x8A, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | <del>Zircaloy</del> (Zr) 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: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)                                      |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)                                       |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 12 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without ~~Zircaloy~~ (Zr) ZR channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | <del>Zircaloy</del> (Zr) 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: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)                                      |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)                                       |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                |

Table 2.1-1 (page 13 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without ~~Zircaloy (Zr)~~ ZR channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the uranium oxide BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

- |  |  |
|--|--|
| a. Cladding Type:  | Zircaloy (Zr) ZR   |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                    | As specified in Table 2.1-3 for the applicable original fuel assembly array/class.                       |
| c. Initial Maximum Rod Enrichment:                               | As specified in Table 2.1-3 for the applicable original fuel assembly array/class.                       |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU for the original fuel assembly. |
| e. Decay Heat Per Assembly                                       | $\leq$ 115 Watts   |
| f. Original Fuel Assembly Length                                 | $\leq$ 135.0 inches (nominal design)   |
| g. Original Fuel Assembly Width                                  | $\leq$ 4.70 inches (nominal design)  |
| h. Fuel Debris Weight  | $\leq$ 550 lbs, including channels and DFC   |



Table 2.1-1 (page 14 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without ~~Zircaloy (Zr)~~ ZR channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | <del>Zircaloy (Zr)</del> ZR   |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 15 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without ~~Zircaloy (Zr)~~ ZR channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | <del>Zircaloy (Zr)</del> ZR   |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                  |

Table 2.1-1 (page 16 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without ~~Zircaloy (Zr)~~ ZR channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the MOX BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | Zircaloy (Zr) ZR   |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.                                   |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.                                   |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM for the original fuel assembly. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts   |
| f. Original Fuel Assembly Length:                                 | $\leq$ 135.0 inches (nominal design)   |
| g. Original Fuel Assembly Width:                                  | $\leq$ 4.70 inches (nominal design)  |
| h. Fuel Debris Weight:  | $\leq$ 550 lbs, including channels and DFC   |

Table 2.1-1 (page 17 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

7. Thoria rods ( $\text{ThO}_2$  and  $\text{UO}_2$ ) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | Zircaloy (Zr) ZR   |
| b. Composition:   | 98.2 wt.% $\text{ThO}_2$ , 1.8 wt. % $\text{UO}_2$ with an enrichment of 93.5 wt. % $^{235}\text{U}$ . |
| c. Number of Rods Per Thoria Rod Canister:  | $\leq 18$  |
| d. Decay Heat Per Thoria Rod Canister:  | $\leq 115$ Watts   |
| e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister: | A fuel post-irradiation cooling time $\geq 18$ years and an average burnup $\leq 16,000$ MWD/MTIHM.    |
| f. Initial Heavy Metal Weight:  | $\leq 27$ kg/canister  |
| g. Fuel Cladding O.D.:  | $\geq 0.412$ inches  |
| h. Fuel Cladding I.D.:  | $\leq 0.362$ inches  |
| i. Fuel Pellet O.D.:  | $\leq 0.358$ inches  |
| j. Active Fuel Length:  | $\leq 111$ inches  |
| k. Canister Weight:   | $\leq 550$ lbs, including fuel   |

Table 2.1-1 (page 18 of 339)  
Fuel Assembly Limits

---

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

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

(All fuel assemblies must be array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A):

Up to four (4) DFCs containing uranium oxide BWR FUEL DEBRIS or MOX BWR FUEL DEBRIS. The remaining MPC-68F fuel storage locations may be filled with fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES;
2. MOX BWR INTACT FUEL ASSEMBLIES;
3. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES placed in DFCs;
4. MOX BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium source material shall be in a water rod location.

Table 2.1-1 (page 19 of 339)  
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

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:   | Zircaloy (Zr) 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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.  |
| ii. All Other Array/Classes                                       | As specified in <i>Section 2.4. Tables 2.1-4 or 2.1-6.</i>  |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.  |

Table 2.1-1 (page 20 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per ~~Assembly~~*Fuel Storage Location*:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 710$  Watts.

ii. All other Array/Classes As specified in *Section 2.4. Tables 2.1-5 or 2.1-7.*

e. Fuel Assembly Length:  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width:  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight:  $\leq 1,680$  lbs (including NON-FUEL HARDWARE)

Table 2.1-1 (page 21 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

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

a. Cladding Type: ~~Zircaloy (Zr)~~ ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class

b. Initial Enrichment: ~~≤ 4.0 wt% <sup>235</sup>U~~. 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  $\geq$  8 years and an average burnup  $\leq$  40,000 MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4. ~~Tables 2.1-4 or 2.1-6.~~

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.



Table 2.1-1 (page 22 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per  
~~Assembly~~ *Fuel Storage*  
*Location:*

- |  |  |
|--|--|
| i. Array/Classes 14x14D,<br>14x14E, and 15x15G | $\leq 710$ Watts.  |
| ii. All Other Array/Classes                    | As specified in <i>Section 2.4. Tables 2.1-5</i><br><del>or 2.1-7.</del> |

- e. Fuel Assembly Length  $\leq 176.8$  inches (nominal design)
- f. Fuel Assembly Width  $\leq 8.54$  inches (nominal design)
- g. Fuel Assembly Weight  $\leq 1,680$  lbs (including NON-FUEL  
HARDWARE and DFC)

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. *Neutron sources and FUEL DEBRIS is* are not authorized for loading in the MPC-24E.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, ~~or~~ orifice rod assemblies, *or vibration suppressor inserts* may be stored in any fuel ~~cell~~ *storage* location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 23 of 339)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32

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:	Zircaloy (Zr) 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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU.
---	--

ii. All Other Array/Classes	As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>
-----------------------------	--

iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.
------------------------	------------------------------

Table 2.1-1 (page 24 of 339)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per <del>Assembly</del> <i>Fuel Storage Location:</i>		
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 500 Watts	
ii. All Other Array/Classes	As specified in <i>Section 2.4. Tables 2.1-5</i> <del>or 2.1-7.</del>	
e. Fuel Assembly Length	≤ 176.8 inches (nominal design)	
f. Fuel Assembly Width	≤ 8.54 inches (nominal design)	
g. Fuel Assembly Weight	≤ 1,680 lbs (including NON-FUEL HARDWARE)	

Table 2.1-1 (page 245 of 339)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES 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  $\geq 9$  years and an average burnup  $\leq 30,000$  MWD/MTU or cooling time  $\geq 20$  years and an average burnup  $\leq 40,000$  MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 226 of 339)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 500$  Watts.

ii. All Other Array/Classes As specified in Section 2.4.

e. Fuel Assembly Length  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight  $\leq 1,680$  lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to eight (8) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32 fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. ~~DAMAGED FUEL ASSEMBLIES~~ and Neutron sources and FUEL DEBRIS are not authorized for loading in the MPC-32.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, or orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 13, 14, 19, and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 257 of 339)  
Fuel Assembly Limits

VI. MPC MODEL: 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:  | Zircaloy (Zr) 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 $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU (or MTU/MTIHM).                          |
| ii. Array/Class 8x8F   | Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.   |
| iii. Array/Classes 10x10D and 10x10E                             | Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.   |
| iv. All Other Array/Classes                                      | As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>  |

Table 2.1-1 (page 268 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Classes 6x6A, 6X6b, 6x6C, 7x7A, and 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 <i>Section 2.4. Tables 2.1-5 or 2.1-7.</i>

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	≤ 700 lbs, including channels and DFC

Table 2.1-1 (page 279 of 339)  
Fuel Assembly Limits

VI. MPC MODEL: 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:   | Zircaloy (Zr) 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                                       | $\leq 4.0$ wt.% <sup>235</sup> 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 $\geq 18$ years and an average burnup $\leq 30,000$ MWD/MTU (or MWD/MTIHM).                              |
| ii. Array/Class 8x8F  | Cooling time $\geq 10$ years and an average burnup $\leq 27,500$ MWD/MTU.   |
| iii. Array/Class 10x10D and 10x10E                                | Cooling time $\geq 10$ years and an average burnup $\leq 22,500$ MWD/MTU.   |
| iv. All Other Array/Classes                                       | As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>  |



Table 2.1-1 (page 28 30 of 339)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

- |      |  |   |
|------|--|---|
| i.   | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 115 Watts  |
| ii.  | Array/Class 8x8F                               | $\leq$ 183.5 Watts  |
| iii. | Array/Classes 10x10D and<br>10x10E             | $\leq$ 95 Watts   |
| iv.  | All Other Array/Classes                        | As specified in <i>Section 2.4. Tables 2.1-5 or 2.1-7</i> |

f. Fuel Assembly Length

- |     |  |                                      |
|-----|--|--------------------------------------|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 135.0 inches (nominal design) |
| ii. | All Other Array/Classes                        | $\leq$ 176.5 inches (nominal design) |

g. Fuel Assembly Width

- |     |  |                                     |
|-----|--|-------------------------------------|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 4.70 inches (nominal design) |
| ii. | All Other Array/Classes                        | $\leq$ 5.85 inches (nominal design) |

h. Fuel Assembly Weight

- |     |  |  |
|-----|--|--|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 550 lbs, including channels and DFC |
| ii. | All Other Array/Classes                        | $\leq$ 700 lbs, including channels and DFC |

Table 2.1-1 (page 29 31 of 339)  
Fuel Assembly limits

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

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

1. For fuel assembly array/classes 6x6A, 6X6B, 6x6C, 7x7A, or 8x8A, up to 68 BWR INTACT FUEL ASSEMBLIES and/or DAMAGED FUEL ASSEMBLIES. Up to eight (8) DFCs containing FUEL DEBRIS from these array/classes may be stored.
2. For all other array/classes, 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 MPC-68FF fuel storage locations may be filled with fuel assemblies of the following type:
  - i. Uranium Oxide BWR INTACT FUEL ASSEMBLIES; or
  - ii. MOX BWR INTACT FUEL ASSEMBLIES.

C. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68FF. The Antimony-Beryllium source material shall be in a water rod location.

D. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

Table 2.1-1 (page 302 of 339)  
Fuel Assembly Limits

VII. MPC MODEL: 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:   | Zircaloy (Zr) 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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.  |
| ii. All Other Array/Classes                                       | As specified in Section 2.4. <del>Tables 2.1-4 or 2.1-6.</del>  |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.  |

Table 2.1-1 (page 343 of 339)  
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

- |   |             |  |
|---|-------------|--|
| d. Decay Heat Per <del>Assembly</del><br><i>Storage Location:</i> | <i>Fuel</i> |  |
| i. Array/Classes 14x14D,<br>14x14E, and 15x15G                    |             | ≤ 710 Watts.   |
| ii. All other Array/Classes                                       |             | As specified in <i>Section 2.4. Tables 2.1-5 or 2.1-7.</i> |
| e. Fuel Assembly Length:  |             | ≤ 176.8 inches (nominal design)                            |
| f. Fuel Assembly Width:   |             | ≤ 8.54 inches (nominal design)                             |
| g. Fuel Assembly Weight:  |             | ≤ 1,680 lbs (including NON-FUEL<br>HARDWARE)               |

Table 2.1-1 (page 324 of 339)  
Fuel Assembly Limits

VII. MPC MODEL: 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: Zircaloy (Zr) ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class

b. Initial Enrichment:  $\leq 4.0$  wt% <sup>235</sup>U. 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  $\geq 8$  years and an average burnup  $\leq 40,000$  MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4. Tables 2.1-4 or 2.1-6.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 335 of 339)  
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

d. Decay Heat Per  
~~Assembly~~ *Fuel Storage*  
*Location:*

- i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 710$  Watts.
- ii. All Other Array/Classes As specified in *Section 2.4. Tables 2.1-5 or 2.1-7.*

- e. Fuel Assembly Length  $\leq 176.8$  inches (nominal design)
- f. Fuel Assembly Width  $\leq 8.54$  inches (nominal design)
- g. Fuel Assembly Weight  $\leq 1,680$  lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24EF fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. *Neutron sources are not authorized for loading in the MPC-24EF.*

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, or orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel ~~cell~~-storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 36 of 39)  
Fuel Assembly Limits

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VIII. MPC MODEL: 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  $\geq$  9 years and an average burnup  $\leq$  30,000 MWD/MTU or cooling time  $\geq$  20 years and an average burnup  $\leq$  40,000 MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 37 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (cont'd)

d. Decay Heat Per Fuel  
Storage Location:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 500$  Watts.

ii. All Other Array/Classes As specified in Section 2.4.

e. Fuel Assembly Length  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight  $\leq 1,680$  lbs (including NON-FUEL  
HARDWARE)



Table 2.1-1 (page 38 of 39)  
Fuel Assembly Limits

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VIII. MPC MODEL: 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  $\geq 9$  years and an average burnup  $\leq 30,000$  MWD/MTU or cooling time  $\geq 20$  years and an average burnup  $\leq 40,000$  MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 39 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (cont'd)

d. Decay Heat Per Fuel  
Storage Location:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 500$  Watts.

ii. All Other Array/Classes As specified in Section 2.4.

e. Fuel Assembly Length  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight  $\leq 1,680$  lbs (including NON-FUEL  
HARDWARE and DFC)

B. Quantity per MPC: Up to eight (8) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32F fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron Sources are not authorized for loading in the MPC-32F.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 13, 14, 19 and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-2 (page 1 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	14x14E
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407 365	≤ 407 412	≤ 425 438	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 5.0 (24E/24EF)	≤ 5.0 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Notes 5 and 7) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Rod Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Rod Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table 2.1-2 (page 2 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR
Design Initial U (kg/assy.) (Note 3)	$\leq 464\ 473$	$\leq 464\ 473$	$\leq 464\ 473$	$\leq 475\ 495$	$\leq 475\ 495$	$\leq 475\ 495$
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)	$\leq 4.1$ (24) $< 4.5$ (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Notes 5 and 7) (wt % <sup>235</sup> U)	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Rod Clad O.D. (in.)	$\geq 0.418$	$\geq 0.420$	$\geq 0.417$	$\geq 0.430$	$\geq 0.428$	$\geq 0.428$
Fuel Rod Clad I.D. (in.)	$\leq 0.3660$	$\leq 0.3736$	$\leq 0.3640$	$\leq 0.3800$	$\leq 0.3790$	$\leq 0.3820$
Fuel Pellet Dia. (in.)	$\leq 0.3580$	$\leq 0.3671$	$\leq 0.3570$	$\leq 0.3735$	$\leq 0.3707$	$\leq 0.3742$
Fuel Rod Pitch (in.)	$\leq 0.550$	$\leq 0.563$	$\leq 0.563$	$\leq 0.568$	$\leq 0.568$	$\leq 0.568$
Active Fuel Length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	$\geq 0.0165$	$\geq 0.015$	$\geq 0.0165$	$\geq 0.0150$	$\geq 0.0140$	$\geq 0.0140$

Table 2.1-2 (page 3 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	15x15G	15x15H	16x16A	17x17A	17x17B	17x17C
Clad Material (Note 2)	SS	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475 495	≤ 443 448	≤ 467 433	≤ 467 474	≤ 474 480
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Notes 5 and 7) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Rod Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Rod Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1-2 (page 4 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. ~~Zr designates cladding material made of zirconium or zirconium alloys~~ Deleted.
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances.
4. Each guide tube replaces four fuel rods.
5. Soluble boron concentration per LCO 3.3.1.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum initial enrichment of the INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS is 4.0 wt.% <sup>235</sup>U.

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

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195 198	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 2.7	≤ 2.7 for the UO <sub>2</sub> rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Rod Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Rod Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1-3 (2 of 5)  
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR
Design Initial U (kg/assy.) (Note 3)	<del>≤ 191</del> 192	<del>≤ 191</del> 190	<del>≤ 191</del> 190	< <del>191</del> 190	≤ 191	<del>≤ 179</del> 180
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Rod Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Rod Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120



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

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR	Zr ZR
Design Initial U (kg/assy.) (Note 3)	<del>≤ 179</del> 180	<del>≤ 179</del> 182	<del>≤ 179</del> 182	<del>≤ 179</del> 183	<del>≤ 179</del> 183	<del>≤ 179</del> 164
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Rod Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Rod Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

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

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	Zr ZR	Zr ZR	Zr ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ <del>188</del> 179	≤ 125	≤ 125
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 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
Fuel Rod Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Rod Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

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. ~~Zr designates cladding material made of zirconium or zirconium alloys 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.  $\leq 0.635$  wt. %  $^{235}\text{U}$  and  $\leq 1.578$  wt. % total fissile plutonium ( $^{239}\text{Pu}$  and  $^{241}\text{Pu}$ ), (wt. % of total fuel weight, i.e.,  $\text{UO}_2$  plus  $\text{PuO}_2$ ).
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 those MPCs 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.%  $^{235}\text{U}$ , as applicable.

Table 2.1-4

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP  
(UNIFORM FUEL LOADING)

Post-irradiation Cooling Time (years)	MPG-24 PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPG-24E/24EF PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPG-24E/24EF PWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)	MPG-32 PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPG-68/68FF BWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPG-68/68FF BWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)
≥ 5	40,600	41,100	39,200	32,200	38,300	36,700
≥ 6	45,000	45,000	43,700	36,500	41,600	39,900
≥ 7	45,900	46,300	44,500	37,500	42,300	40,700
≥ 8	48,300	48,900	46,900	39,900	44,800	42,900
≥ 9	50,300	50,700	48,700	41,500	46,600	44,700
≥ 10	51,600	52,100	50,100	42,900	48,000	46,100
≥ 11	53,100	53,700	51,500	44,100	49,600	47,200
≥ 12	54,500	55,100	52,600	45,000	50,800	48,500
≥ 13	55,600	56,100	53,800	45,700	51,800	49,800
≥ 14	56,500	57,100	54,900	46,500	52,700	50,700
≥ 15	57,400	58,000	55,800	47,200	53,900	51,500

Notes: 1. Linear interpolation between points is permitted.

2. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.

Table 2.1-5

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT  
(UNIFORM FUEL LOADING)

Post-irradiation Cooling Time (years)	MPG-24 PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPG-24E/24EF PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPG-24E/24EF PWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)	MPG-32 PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPG-68/68FF BWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPG-68/68FF BWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)
≥ 5	4157	4173	4115	898	414	393
≥ 6	4123	4138	4081	873	394	374
≥ 7	4030	4043	991	805	363	345
≥ 8	4020	4033	981	800	360	342
≥ 9	4010	4023	972	794	358	340
≥ 10	4000	4012	962	789	355	337
≥ 11	996	4008	958	785	353	336
≥ 12	992	4004	954	782	352	334
≥ 13	987	999	949	773	350	332
≥ 14	983	995	945	769	348	331
≥ 15	979	991	941	766	347	329

Notes: 1. Linear interpolation between points is permitted.

2. Includes all sources of heat (i.e., fuel and NON-FUEL HARDWARE).

Table 2.1-6 (page 1 of 2)

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP  
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 2 (MWD/MTU)
≥5	49,800	32,200	51,600	32,200
≥6	56,100	37,400	58,400	37,400
≥7	56,400	41,100	58,500	41,100
≥8	58,800	43,800	60,900	43,800
≥9	60,400	45,800	62,300	45,800
≥10	61,200	47,500	63,300	47,500
≥11	62,400	49,000	64,900	49,000
≥12	63,700	50,400	65,900	50,400
≥13	64,800	51,500	66,800	51,500
≥14	65,500	52,500	67,500	52,500
≥15	66,200	53,700	68,200	53,700
≥16	-	55,000	-	55,000
≥17	-	55,900	-	55,900
≥18	-	56,800	-	56,800
≥19	-	57,800	-	57,800
≥20	-	58,800	-	58,800

- Notes: 1. Linear interpolation between points is permitted.  
 2. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.  
 3. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.

Table 2.1-6 (page 2 of 2)

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP  
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-32 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-32 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	39,800	22,100	45,100	26,200
≥ 6	43,400	26,200	47,400	30,500
≥ 7	44,500	29,100	47,400	33,600
≥ 8	46,700	31,200	50,400	35,900
≥ 9	48,400	32,700	52,100	37,600
≥ 10	49,600	34,100	53,900	39,000
≥ 11	50,900	35,200	55,500	40,200
≥ 12	51,900	36,200	56,500	41,200
≥ 13	52,900	37,000	57,500	42,300
≥ 14	53,800	37,800	58,800	43,300
≥ 15	54,700	38,600	59,900	44,200
≥ 16	-	39,400	-	45,000
≥ 17	-	40,200	-	45,900
≥ 18	-	40,800	-	46,700
≥ 19	-	41,500	-	47,500
≥ 20	-	42,200	-	48,500

- Notes: 1. Linear interpolation between points is permitted.  
 2. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.  
 3. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.

Table 2.1-7 (page 1 of 2)

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT  
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Decay Heat for Region 1 (Watts)	MPC-24 PWR Assembly Decay Heat for Region 2 (Watts)	MPC-24E/24EF PWR Assembly Decay Heat for Region 1 (Watts)	MPC-24E/24EF PWR Assembly Decay Heat for Region 2 (Watts)
≥5	1470	900	1540	900
≥6	1470	900	1540	900
≥7	1335	900	1395	900
≥8	1,301	900	1360	900
≥9	1268	900	1325	900
≥10	1235	900	1290	900
≥11	1224	900	1275	900
≥12	1207	900	1260	900
≥13	1193	900	1245	900
≥14	1179	900	1230	900
≥15	1165	900	1215	900
≥16	-	900	-	900
≥17	-	900	-	900
≥18	-	900	-	900
≥19	-	900	-	900
≥20	-	900	-	900

- Notes: 1. Linear interpolation between points is permitted.  
 2. Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).  
 4. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.



Table 2.1-7 (page 2 of 2)

TABLE DELETED

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT  
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-32 PWR Assembly Decay Heat for Region 1 (Watts)	MPC-32 PWR Assembly Decay Heat for Region 2 (Watts)	MPC-68/68FF BWR Assembly Decay Heat for Region 1 (Watts)	MPC-68/68FF BWR Assembly Decay Heat for Region 2 (Watts)
≥5	1131	600	500	275
≥6	1072	600	468	275
≥7	993	600	418	275
≥8	978	600	414	275
≥9	964	600	410	275
≥10	950	600	405	275
≥11	943	600	403	275
≥12	937	600	400	275
≥13	931	600	397	275
≥14	924	600	394	275
≥15	918	600	391	275
≥16	-	600	-	275
≥17	=	600	=	275
≥18	=	600	=	275
≥19	-	600	-	275
≥20	-	600	-	275

- Notes: 1. Linear interpolation between points is permitted.  
 2. Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).  
 3. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.

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

Post-irradiation Cooling Time (years)	NEUTRON POISON INSERTS BURNUP (MWD/MTU) (Note 34)	GUIDE TUBE HARDWARE BURNUP (MWD/MTU) (Note 45)	CONTROL COMPONENT BURNUP (MWD/MTU) (Note 56)	APSR BURNUP (MWD/MTU)
≥ 3	≤ <del>20,000</del> 24,635	NA (Note 67)	NA	NA
≥ 4	≤ <del>25,000</del> 30,000	≤ 20,000	NA	NA
≥ 5	≤ <del>30,000</del> 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ <del>40,000</del> 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ <del>45,000</del> 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ <del>50,000</del> 60,000	≤ 45,000	-	≤ 83,000
≥ 9	≤ <del>60,000</del>	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 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 TPD 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), and Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts.
  5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
  6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
  7. NA means not authorized for loading at this cooling time.

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. A detailed discussion of how 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)

<b>MPC Model</b>	<b>Decay Heat per Fuel Storage Location (kW)</b>
<i>Intact Fuel Assemblies</i>	
MPC-24	$\leq 1.157$
MPC-24E/24EF	$\leq 1.173$
MPC-32/32F	$\leq 0.898$
MPC-68/68FF	$\leq 0.414$
<i>Damaged Fuel Assemblies and Fuel Debris</i>	
MPC-24	$\leq 1.099$
MPC-24E/24EF	$\leq 1.114$
MPC-32/32F	$\leq 0.718$
MPC-68/68FF	$\leq 0.393$

2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.4-2 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in regionalized loading for each MPC model.

2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel (cont'd)

Table 2.4-2

Fuel Storage Regions and Maximum Decay Heat per MPC

<b>MPC Model</b>	<b>Number of Fuel Storage Locations in Inner and Outer Regions</b>	<b>Inner Region Maximum Decay Heat per Assembly (kW)</b>	<b>Outer Region Maximum Decay Heat per Assembly (kW)</b>
MPC-24	4 and 20	1.470	0.900
MPC-24E/24EF	4 and 20	1.540	0.900
MPC-32/32F	12 and 20	1.131	0.600
MPC-68/68FF	32 and 36	0.500	0.275

2.4.3 Burnup 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.

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

Equation 2.4.3

Where:

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

### 2.4.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel (cont'd)

$q$  = 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. %  $^{235}\text{U}$ )  
(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	20277.1	303.592	-68.329	-139.41	2993.67	-498.159	-615.411
≥ 4	35560.1	-6034.67	985.415	-132.734	3578.92	-723.721	-609.84
≥ 5	48917.9	-14499.5	2976.09	-150.707	4072.55	-892.691	-54.8362
≥ 6	59110.3	-22507	5255.61	-177.017	4517.03	-1024.01	613.36
≥ 7	67595.6	-30158.1	7746.6	-200.128	4898.71	-1123.21	716.004
≥ 8	74424.9	-36871.1	10169.4	-218.676	5203.64	-1190.24	741.163
≥ 9	81405.8	-44093.1	12910.8	-227.916	5405.34	-1223.27	250.224
≥ 10	86184.3	-49211.7	15063.4	-237.641	5607.96	-1266.21	134.435
≥ 11	92024.9	-55666.8	17779.6	-240.973	5732.25	-1282.12	-401.456
≥ 12	94775.8	-58559.7	19249.9	-246.369	5896.27	-1345.42	-295.435
≥ 13	100163	-64813.8	22045.1	-242.572	5861.86	-1261.66	-842.159
≥ 14	103971	-69171	24207	-242.651	5933.96	-1277.48	-1108.99
≥ 15	108919	-75171.1	27152.4	-243.154	6000.2	-1301.19	-1620.63
≥ 16	110622	-76715.2	28210.2	-240.235	6028.33	-1307.74	-1425.5
≥ 17	115582	-82929.7	31411.9	-235.234	5982.3	-1244.11	-1948.05
≥ 18	119195	-87323.5	33881.4	-233.28	6002.43	-1245.95	-2199.41
≥ 19	121882	-90270.6	35713.7	-231.873	6044.42	-1284.55	-2264.05
≥ 20	124649	-93573.5	37853.1	-230.22	6075.82	-1306.57	-2319.63

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	18937.9	70.2997	-28.6224	-130.732	2572.36	-383.393	-858.17
≥ 4	32058.7	-4960.63	745.224	-125.978	3048.98	-551.656	-549.108
≥ 5	42626.3	-10804.1	1965.09	-139.722	3433.49	-676.643	321.88
≥ 6	51209.6	-16782.3	3490.45	-158.929	3751.01	-761.524	847.282
≥ 7	57829.9	-21982	5009.12	-180.026	4066.65	-846.272	1200.45
≥ 8	62758	-26055.3	6330.88	-196.804	4340.18	-928.336	1413.17
≥ 9	68161.4	-30827.6	7943.87	-204.454	4500.52	-966.347	1084.69
≥ 10	71996.8	-34224.3	9197.25	-210.433	4638.94	-1001.83	1016.38
≥ 11	75567.3	-37486.1	10466.9	-214.95	4759.55	-1040.85	848.169
≥ 12	79296.7	-40900.3	11799.6	-212.898	4794.13	-1040.51	576.242
≥ 13	82257.3	-43594	12935	-212.8	4845.81	-1056.01	410.807
≥ 14	83941.2	-44915.2	13641	-215.389	4953.19	-1121.71	552.724
≥ 15	87228.5	-48130	15056.9	-212.545	4951.12	-1112.5	260.194
≥ 16	90321.7	-50918.3	16285.5	-206.094	4923.36	-1106.35	-38.7487
≥ 17	92836.2	-53314.5	17481.7	-203.139	4924.61	-1109.32	-159.673
≥ 18	93872.8	-53721.4	17865.1	-202.573	4956.21	-1136.9	30.0594
≥ 19	96361.6	-56019.1	19075.9	-199.068	4954.59	-1156.07	-125.917
≥ 20	98647.5	-57795.1	19961.8	-191.502	4869.59	-1108.74	-217.603

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	19176.9	192.012	-66.7595	-138.112	2666.73	-407.664	-1372.41
≥ 4	32040.3	-4731.4	651.014	-124.944	3012.63	-530.456	-890.059
≥ 5	43276.7	-11292.8	2009.76	-142.172	3313.91	-594.917	-200.195
≥ 6	51315.5	-16920.5	3414.76	-164.287	3610.77	-652.118	463.041
≥ 7	57594.7	-21897.6	4848.49	-189.606	3940.67	-729.367	781.46
≥ 8	63252.3	-26562.8	6273.01	-199.974	4088.41	-732.054	693.879
≥ 9	67657.5	-30350.9	7533.4	-211.77	4283.39	-772.916	588.456
≥ 10	71834.4	-34113.7	8857.32	-216.408	4383.45	-774.982	380.243
≥ 11	75464.1	-37382.1	10063	-218.813	4460.69	-776.665	160.668
≥ 12	77811.1	-39425.1	10934.3	-225.193	4604.68	-833.459	182.463
≥ 13	81438.3	-42785.4	12239.9	-220.943	4597.28	-803.32	-191.636
≥ 14	84222.1	-45291.6	13287.9	-218.366	4608.13	-791.655	-354.59
≥ 15	86700.1	-47582.6	14331.2	-218.206	4655.34	-807.366	-487.316
≥ 16	88104.7	-48601.1	14927.9	-219.498	4729.97	-849.446	-373.196
≥ 17	91103.3	-51332.5	16129	-212.138	4679.91	-822.896	-654.296
≥ 18	93850.4	-53915.8	17336.9	-207.666	4652.65	-799.697	-866.307
≥ 19	96192.9	-55955.8	18359.3	-203.462	4642.65	-800.315	-1007.75
≥ 20	97790.4	-57058.1	19027.7	-200.963	4635.88	-799.721	-951.122



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	15789.2	119.829	-21.8071	-127.422	2152.53	-267.717	-580.768
≥ 4	26803.8	-3312.93	415.027	-116.279	2550.15	-386.33	-367.168
≥ 5	36403.6	-7831.93	1219.66	-126.065	2858.32	-471.785	326.863
≥ 6	44046.1	-12375.9	2213.52	-145.727	3153.45	-539.715	851.971
≥ 7	49753.5	-16172.6	3163.61	-166.946	3428.38	-603.598	1186.31
≥ 8	55095.4	-20182.5	4287.03	-183.047	3650.42	-652.92	1052.4
≥ 9	58974.4	-23071.6	5156.53	-191.718	3805.41	-687.18	1025
≥ 10	62591.8	-25800.8	5995.95	-195.105	3884.14	-690.659	868.556
≥ 11	65133.1	-27747.4	6689	-203.095	4036.91	-744.034	894.607
≥ 12	68448.4	-30456	7624.9	-202.201	4083.52	-753.391	577.914
≥ 13	71084.4	-32536.4	8381.78	-201.624	4117.93	-757.16	379.105
≥ 14	73459.5	-34352.3	9068.86	-197.988	4113.16	-747.015	266.536
≥ 15	75950.7	-36469.4	9920.52	-199.791	4184.91	-779.222	57.9429
≥ 16	76929.1	-36845.6	10171.3	-197.88	4206.24	-794.541	256.099
≥ 17	79730	-39134.8	11069.4	-190.865	4160.42	-773.448	-42.6853
≥ 18	81649.2	-40583	11736.1	-187.604	4163.36	-785.838	-113.614
≥ 19	83459	-41771.8	12265.9	-181.461	4107.51	-758.496	-193.442
≥ 20	86165.4	-44208.8	13361.2	-178.89	4107.62	-768.671	-479.778

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						
	A	B	C	D	E	F	G
≥ 3	15192.5	50.5722	-12.3042	-126.906	2009.71	-235.879	-561.574
≥ 4	25782.5	-3096.5	369.096	-113.289	2357.75	-334.695	-254.964
≥ 5	35026.5	-7299.87	1091.93	-124.619	2664	-414.527	470.916
≥ 6	42234.9	-11438.4	1967.63	-145.948	2945.81	-474.981	1016.84
≥ 7	47818.4	-15047	2839.22	-167.273	3208.95	-531.296	1321.12
≥ 8	52730.7	-18387.2	3702.43	-175.057	3335.58	-543.232	1223.61
≥ 9	56254.6	-20999.9	4485.93	-190.489	3547.98	-600.64	1261.55
≥ 10	59874.6	-23706.5	5303.88	-193.807	3633.01	-611.892	1028.63
≥ 11	62811	-25848.4	5979.64	-194.997	3694.14	-618.968	862.738
≥ 12	65557.6	-27952.4	6686.74	-198.224	3767.28	-635.126	645.139
≥ 13	67379.4	-29239.2	7197.49	-200.164	3858.53	-677.958	652.601
≥ 14	69599.2	-30823.8	7768.51	-196.788	3868.2	-679.88	504.443
≥ 15	71806.7	-32425	8360.38	-191.935	3851.65	-669.917	321.146
≥ 16	73662.6	-33703.5	8870.78	-187.366	3831.59	-658.419	232.335
≥ 17	76219.8	-35898.1	9754.72	-189.111	3892.07	-694.244	-46.924
≥ 18	76594.4	-35518.2	9719.78	-185.11	3897.04	-712.82	236.047
≥ 19	78592.7	-36920.8	10316.5	-179.54	3865.84	-709.551	82.478
≥ 20	80770.5	-38599.9	11051.3	-175.106	3858.67	-723.211	-116.014

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	17038.2	158.445	-37.6008	-136.707	2368.1	-321.58	-700.033
≥ 4	29166.3	-3919.95	508.439	-125.131	2782.53	-455.722	-344.199
≥ 5	40285	-9762.36	1629.72	-139.652	3111.83	-539.804	139.67
≥ 6	48335.7	-15002.6	2864.09	-164.702	3444.97	-614.756	851.706
≥ 7	55274.9	-20190	4258.03	-185.909	3728.11	-670.841	920.035
≥ 8	60646.6	-24402.4	5483.54	-199.014	3903.29	-682.26	944.913
≥ 9	64663.2	-27753.1	6588.21	-215.318	4145.34	-746.822	967.914
≥ 10	69306.9	-31739.1	7892.13	-218.898	4237.04	-746.815	589.277
≥ 11	72725.8	-34676.6	8942.26	-220.836	4312.93	-750.85	407.133
≥ 12	76573.8	-38238.7	10248.1	-224.934	4395.85	-757.914	23.7549
≥ 13	78569	-39794.3	10914.9	-224.584	4457	-776.876	69.428
≥ 14	81559.4	-42453.6	11969.6	-222.704	4485.28	-778.427	-203.031
≥ 15	84108.6	-44680.4	12897.8	-218.387	4460	-746.756	-329.078
≥ 16	86512.2	-46766.8	13822.8	-216.278	4487.79	-759.882	-479.729
≥ 17	87526.7	-47326.2	14221	-218.894	4567.68	-805.659	-273.692
≥ 18	90340.3	-49888.6	15349.8	-212.139	4506.29	-762.236	-513.316
≥ 19	93218.2	-52436.7	16482.4	-207.653	4504.12	-776.489	-837.1
≥ 20	95533.9	-54474.1	17484.2	-203.094	4476.21	-760.482	-955.662

Table 2.4-3 (Page 7 of 8)

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

Cooling Time (years)	Array/Class 17x17A						
	A	B	C	D	E	F	G
≥ 3	16784.4	3.90244	-10.476	-128.835	2256.98	-287.108	-263.081
≥ 4	28859	-3824.72	491.016	-120.108	2737.65	-432.361	-113.457
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

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	15526.8	18.0364	-9.36581	-128.415	2050.81	-243.915	-426.07
≥ 4	26595.4	-3345.47	409.264	-115.394	2429.48	-350.883	-243.477
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953

Table 2.4-4 (Page 1 of 10)

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

Cooling Time (years)	Array/Class 7x7B						
	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						
	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

### 3.0 DESIGN FEATURES

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#### 3.1 Site

##### 3.1.1 Site Location

The HI-STORM 100 Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

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#### 3.2 Design Features Important for Criticality Control

##### 3.2.1 MPC-24

1. Flux trap size:  $\geq 1.09$  in.
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.0267$  g/cm<sup>2</sup> (Boral) or  $\geq 0.0223$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.2 MPC-68 and MPC-68FF

1. Fuel cell pitch:  $\geq 6.43$  in.
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) or  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.3 MPC-68F

1. Fuel cell pitch:  $\geq 6.43$  in.
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.01$  g/cm<sup>2</sup>

##### 3.2.4 MPC-24E and MPC-24EF

1. Flux trap size:
  - i. Cells 3, 6, 19, and 22:  $\geq 0.776$  inch
  - ii. All Other Cells:  $\geq 1.076$  inches
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) or  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.5 MPC-32 and MPC-32F

1. Fuel cell pitch:  $\geq 9.158$  inches
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) or  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |



## DESIGN FEATURES

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### 3.2 Design Features Important for Criticality Control (con't)

3.2.6 ~~Fuel spacers shall be sized~~ *The fuel assemblies shall be positioned in the MPC to ensure that the active fuel region of intact fuel assemblies remains within the Boraf neutron poison region of the MPC basket with water in the MPC.*

3.2.7 *The  $B_4C$  content in METAMIC shall be  $\leq 33.0$  wt.%.*

#### 3.2.8 Neutron Absorber Tests

*Section 9.1.5.3 of the HI-STORM 100 FSAR is hereby incorporated by reference into the HI-STORM 100 CoC. The minimum  $^{10}B$  for the neutron absorber shall meet the minimum requirements for each MPC model specified in Sections 3.2.1 through 3.2.5 above.*

### 3.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing Code for the HI-STORM 100 System, as clarified in Specification 3.3.1 below, *except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections provided a written reconciliation of the later edition against the 1995 Edition, including addenda, is performed by the certificate holder.* American Concrete Institute (ACI) 349-85 is the governing Code for plain concrete as clarified in Appendix 1.D of the Final Safety Analysis Report for the HI-STORM 100 Cask System.

#### 3.3.1 ~~Exceptions~~ Alternatives to Codes, Standards, and Criteria

Table 3-1 lists approved ~~exceptions~~ *alternatives* to the ASME Code for the design of the HI-STORM 100 Cask System.

#### 3.3.2 Construction/Fabrication ~~Exceptions~~ Alternatives to Codes, Standards, and Criteria

Proposed alternatives to the ASME Code, Section III, 1995 Edition with Addenda through 1997 including ~~exceptions~~ *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

(continued)

DESIGN FEATURES

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3.3.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria (cont'd)

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 ~~exceptions~~ *alternatives* shall be submitted in accordance with 10 CFR 72.4.

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(continued)

DESIGN FEATURES

Table 3-1 (page 1 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

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

Table 3-1 (page 2 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
<p>MPC basket supports and lift lugs</p>	<p>NB-1130</p>	<p>NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than <math>2t</math> from the pressure-retaining portion of the component, where <math>t</math> is the nominal thickness of the pressure-retaining material.</p> <p>NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within <math>2t</math> from the pressure-retaining portion of the component.</p>	<p>The MPC basket supports (nonpressure-retaining structural attachments) and lift lugs (nonstructural attachments (relative to the function of lifting a loaded MPC) that are used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.</p>

Table 3-1 (page 3 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
<i>MPC, MPC basket assembly, HI-STORM OVERPACK and HI-TRAC TRANSFER CASK</i>	<i>NB-3100 NG-3100 NF-3100</i>	<i>Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.</i>	<i>These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.</i>
MPC	NB-3350	<i>NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.</i>	<p><i>Due to MPC basket-to-shell interface requirements, the MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.</i></p> <p><i>From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.</i></p>

Table 3-1 (page 4 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
MPC, MPC Basket Assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	NB-4120 NG-4120 NF-4120	NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.	<p><i>In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of lead are not, unless explicitly stated by the Code, defined as heat treatment operations.</i></p> <p><i>For the steel parts in the HI-STORM 100 System components, the duration for which a part exceeds the off-normal temperature limit defined in Chapter 2 of the FSAR shall be limited to 24 hours in a particular manufacturing process (such as the HI-TRAC lead pouring process).</i></p>
MPC, MPC basket assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	NB-4220 NF-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	<p><i>The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-TRANSFER CASK) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.</i></p>
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3).	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.

Table 3-1 (page 5 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. <del>The MPC vent and drain cover plate welds are leak tested.</del> The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	<p>The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be <del>hydrostatically</del> <i>pressure</i> tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant <del>hydrostatic</del> <i>pressure</i> test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded.</p> <p>The inspection <del>process results</del>, including <i>relevant</i> findings (indications), shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate <del>weld is confirmed by leakage testing and liquid penetrant examination</del> and the closure ring welds <del>is</del> <i>are</i> confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.</p>

Table 3-1 (page 6 of 59)  
LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. The function of the MPC enclosure vessel is to contain the radioactive contents under normal, off-normal, and accident conditions. The MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NG-2000 requirements.



<b>Table 3-1 (page 7 of 59)</b> <b>LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM</b>			
<b>Component</b>	<b>Reference ASME Code Section/Article</b>	<b>Code Requirement</b>	<b>Exception/Alternative, Justification &amp; Compensatory Measures</b>
MPC basket assembly	NG-4420	NG-4427(a) allows a fillet weld in any single continuous weld to be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", $F$ , in the ASME Code for which specific stress intensity limits do not apply).
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. The MPC basket data package to be in accordance with Holtec approved QA program.
OVERPACK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.

Table 3-1 (page 8 of 59) LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM			
Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
TRANSFER CASK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.
OVERPACK Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member of thickness 1 inch or greater is loaded to transmit loads in the through thickness direction.	The <del>large</del> margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The OVERPACK baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have <del>a minimum</del> <i>large</i> safety factors of <del>&gt; 12</del> during lifting. Likewise, <del>the</del> top lid plate to lid shell weld has a <i>large structural margin under the inertia loads imposed during a non-mechanistic tipover event. safety factor &gt; 6</i> under a deceleration of 45 g's.
OVERPACK Steel Structure	NF-3256 NF-3266	Provides requirements for welded joints.	Welds for which no structural credit is taken are identified as "Non-NF" welds in the design drawings by an <del>an</del> <i>"*"</i> . These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX.  <i>Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.</i>

Table 3-1 (page 9 of 59) LIST OF ASME CODE EXCEPTIONS/ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM			
Component	Reference ASME Code Section/Article	Code Requirement	Exception/Alternative, Justification & Compensatory Measures
HI-STORM OVERPACK and HI-TRAC TRANSFER CASK	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting	<p><i>These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The OVERPACK and TRANSFER CASK bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).</i></p> <p><i>Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.</i></p>

DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

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

1. The temperature of 80° F is the maximum average yearly temperature.
2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.
3. a. For free-standing casks, the resultant horizontal acceleration (vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site),  $G_H$ , and vertical ZPA,  $G_V$ , expressed as fractions of 'g', shall satisfy the following inequality:

$$G_H + \mu G_V \leq \mu$$

where  $\mu$  is *either* the Coulomb friction coefficient for the HI-STORM 100/ISFSI pad interface *or the ratio*  $r/h$ , where 'r' is the radius of the cask and 'h' is the height of the cask center-of-gravity above the ISFSI pad surface. *The above inequality must be met for both definitions of  $\mu$ . Unless demonstrated by appropriate testing that a higher coefficient of friction value of  $\mu$  is appropriate for a specific ISFSI, the value of  $\mu$  used shall be 0.53. Representative values of  $G_H$  and  $G_V$  combinations for a coefficient of friction = 0.53 to prevent sliding are provided in Table 3-2. If acceleration time histories on the ISFSI pad surface are available,  $G_H$  and  $G_V$  may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality shall be evaluated at each time step in the acceleration time history over the total duration of the seismic event.*

Table 3-2

Representative DBE Acceleration Values to Prevent HI-STORM 100 Sliding ( $\mu = 0.53$ )

Equivalent Vectorial Sum of Two Horizontal ZPA's ( $G_H$ in g's)	Corresponding Vertical ZPA ( $G_V$ in g's)
0.445	0.160
0.424	0.200
0.397	0.250

(continued)

DESIGN FEATURES

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3.4 Site-Specific Parameters and Analyses (continued)

b. *For free-standing casks, under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will not result in cask tipover or cause a cask to fall off the pad. In addition, impact between casks should be precluded, or should be considered an accident for which the maximum g-load experienced by the stored fuel shall be limited to 45 g's.*

c. For those ISFSI sites with design basis seismic acceleration values higher than those allowed for free-standing casks, the HI-STORM 100 System shall be anchored to the ISFSI pad. The site seismic characteristics and the anchorage system shall meet the following requirements:

i. The site acceleration response spectra at the top of the ISFSI pad shall have ZPAs that meet the following inequalities:

$$G_H \leq 2.12$$

AND

$$G_V \leq 1.5$$

Where:

$G_H$  is the vectorial sum of the two horizontal ZPAs at a three-dimensional seismic site (or the horizontal ZPA at a two-dimensional site) and  $G_V$  is the vertical ZPA.

ii. Each HI-STORM 100 dry storage cask shall be anchored with twenty-eight (28), 2-inch diameter studs and compatible nuts of material suitable for the expected ISFSI environment. The studs shall meet the following requirements:

Yield Strength at Ambient Temperature:  $\geq 80$  ksi

Ultimate Strength at Ambient Temperature:  $\geq 125$  ksi

Initial Tensile Pre-Stress:  $\geq 55$  ksi AND  $\leq 65$  ksi

NOTE: The above anchorage specifications are required for the seismic

(continued)

## DESIGN FEATURES

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### 3.4 Site-Specific Parameters and Analyses (continued)

spectra defined in item 3.4.3.b.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:  $\geq 4,000$  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 shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.
- 6.
  - a. For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to  $\leq 45$  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 lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having 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  $\leq 45$  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 lift height above the ISFSI pad is not required to be established if the cask is lifted with a device design in accordance with ANSI N14.6 and having redundant drop protection features.
- 7. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

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(continued)

## DESIGN FEATURES

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### 3.4 Site-Specific Parameters and Analyses (continued)

8. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures  $\geq 0^{\circ}$  F.
9. For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e, longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.

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(continued)

## DESIGN FEATURES

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### 3.5 Cask Transfer Facility (CTF)

#### 3.5.1 TRANSFER CASK and MPC Lifters

Lifting of a loaded TRANSFER CASK and MPC ~~outside~~ *using devices that are not integral to* structures governed by 10 CFR Part 50 shall be performed with a CTF that is designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and the below clarifications. The CTF Structure requirements below do not apply to heavy loads bounded by the regulations of 10 CFR Part 50.

#### 3.5.2 CTF Structure Requirements

##### 3.5.2.1 Cask Transfer Station and Stationary Lifting Devices

1. The metal weldment structure of the CTF structure shall be designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. The applicable loads, load combinations, and associated service condition definitions are provided in Table 3-3. All compression loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF.
2. If a portion of the CTF structure is constructed of reinforced concrete, then the factored load combinations set forth in ACI-318 (89) for the loads defined in Table 3-3 shall apply.
3. The TRANSFER CASK and MPC lifting device used with the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1.
4. The CTF shall be designed, constructed, and evaluated to ensure that if the MPC is dropped during inter-cask transfer operations, its confinement boundary would not be breached. This requirements applies to CTFs with either stationary or mobile lifting devices.

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DESIGN FEATURES

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3.5.2.2 Mobile Lift Devices

If a mobile lifting device is used as the lifting device, in lieu of a stationary lifting device, is shall meet the guidelines of NUREG- 0612, Section 5.1, with the following clarifications:

1. Mobile lifting devices shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6(1)(a) and shall be capable of stopping and holding the load during a Design Basis Earthquake (DBE) event.
2. Mobile lifting devices shall conform to meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes."
3. Mobile cranes are not required to meet the requirements of NUREG-0612, Section 5.1.6(2) for new cranes.
4. Horizontal movements of the TRANSFER CASK and MPC using a mobile crane are prohibited.

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(continued)

DESIGN FEATURES

Table 3-3

Load Combinations and Service Condition Definitions for the CTF Structure (Note 1)

Load Combination	ASME III Service Condition for Definition of Allowable Stress	Comment
D* D + S	Level A	All primary load bearing members must satisfy Level A stress limits
D + M + W' (Note 2) D + F D + E D + Y	Level D	Factor of safety against overturning shall be $\geq 1.1$

D = Dead load  
D\* = Apparent dead load  
S = Snow and ice load for the CTF site  
M = Tornado missile load for the CTF site  
W' = Tornado wind load for the CTF site  
F = Flood load for the CTF site  
E = Seismic load for the CTF site  
Y = Tsunami load for the CTF site

- Notes:
1. The reinforced concrete portion of the CTF structure shall also meet the factored combinations of loads set forth in ACI-318(89).
  2. Tornado missile load may be reduced or eliminated based on a PRA for the CTF site.

## DESIGN FEATURES

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### 3.6 Forced Helium Dehydration System

#### 3.6.1 System Description

Use of the Forced Helium Dehydration (FHD) system, (a closed-loop system) is an alternative to vacuum drying the MPC for moderate burnup fuel ( $\leq 45,000$  MWD/MTU) and mandatory for drying MPCs containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.6.2.

#### 3.6.2 Design Criteria

3.6.2.1 The temperature of the helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.

3.6.2.2 The pressure in the MPC cavity space shall be  $\leq 60.3$  psig (75 psia).

3.6.2.3 The hourly recirculation rate of helium shall be  $\geq 10$  times the nominal helium mass backfilled into the MPC for fuel storage operations.

3.6.2.4 The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. *The limit is met if the helium gas temperature at the demister outlet is verified by measurement to remain  $\leq 21^\circ\text{F}$  for a period of 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain  $\leq 22.9^\circ\text{F}$  for  $\geq 30$  minutes .*

3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point  $\leq 120^\circ\text{F}$ .

3.6.2.6 The demister module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in FSAR Appendix 2.B) has been completed.

3.6.2.7 The helium circulator shall be sized to effect the minimum flow rate of circulation required by these design criteria.

3.6.2.8 The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets these design criteria.

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(continued)

## DESIGN FEATURES

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### 3.6 Forced Helium Dehydration System (continued)

#### 3.6.3 Fuel Cladding Temperature

A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR *Section 4.4, Subsections 4.4.1.1.1 through 4.4.1.1.4*, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation, is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

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**LAR 1014-2, REVISION 2**

**ATTACHMENT 3**

**COC CHANGES – REVISED VERSION**

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**

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

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1014	05/31/00	06/01/20	72-1014	2		USA/72-1014

Issued To: (Name/Address)  
  
Holtec International  
Holtec Center  
555 Lincoln Drive West  
Marlton, NJ 08053

Safety Analysis Report Title  
  
Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System  
Docket No. 72-1014

**CONDITIONS**

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

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 (PWR), fuel assemblies or 68 boiling water reactor (BWR) 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. The cask comprises three discrete components: the MPCs, 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. It is made entirely of stainless steel except for the neutron absorbers and aluminum heat conduction elements (AHCEs), which are installed in some early-vintage MPCs. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. The honeycombed basket, which is equipped with neutron absorbers, provides criticality control.

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

1. b. Description (continued)

There are eight types of MPCs: the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, and MPC-68FF. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. All eight 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 water 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 is the maximum 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 lead and water shielding and larger outer dimensions than the 100 ton HI-TRAC transfer cask.

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 a minimum of four air inlets at the bottom and a minimum of four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has channels attached to its interior surface to guide the MPC during insertion and removal, provide a flexible 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 is a variant of the HI-STORM 100 family and is outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the concrete storage pad in high seismic applications. The HI-STORM 100A applies to both the HI-STORM 100 and HI-STORM 100S overpacks that are classified as the HI-STORM 100A and HI-STORM 100SA, respectively.

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.

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.

**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

5. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC transfer cask, or a HI-STORM 100 or 100S 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 regulatory 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/or Sections 3.4.6 and Section 3.5 of Appendix B to this certificate, as applicable.

6. APPROVED CONTENTS

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B 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 Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM Cask Systems (for each thermally unique MPC basket design - MPC-24/24E/24EF, MPC-32/32F, and MPC-68/68F/68FF) placed into service by any user with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the FSAR.

Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.

Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.



**CERTIFICATE OF COMPLIANCE  
FOR SPENT FUEL STORAGE CASKS**  
Supplemental Sheet

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.
- 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.
- 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.)
- a. Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.
- h. Transfer of the MPC from the transfer cask to the overpack.
- i. Placement of the HI-STORM 100 Cask System at the ISFSI.
- j. HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

11. 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, and the attached Appendix A and Appendix B. 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 different CoC amendments, may be used with one another provided the CoC does not specifically prohibit their use with each other, and a design compatibility assessment is performed by the CoC holder.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director  
Spent Fuel Project Office  
Office of Nuclear Material Safety  
and Safeguards

Attachments:

1. Appendix A
2. Appendix B

**CERTIFICATE OF COMPLIANCE NO. 1014**

**APPENDIX A**

**TECHNICAL SPECIFICATIONS  
FOR THE HI-STORM 100 CASK SYSTEM**

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3.1.2	SFSC Heat Removal System .....	3.1.2-1
3.1.3	Fuel Cool-Down.....	3.1.3-1
3.1.4	Supplemental Cooling System.....	3.1.4-1
3.2	SFSC RADIATION PROTECTION.....	3.2.1-1
3.2.1	Deleted.....	3.2.1-1
3.2.2	TRANSFER CASK SURFACE CONTAMINATION.....	3.2.2-1
3.2.3	Deleted.....	3.2.3-1
3.3	SFSC CRITICALITY CONTROL .....	3.3-1
3.3.1	Boron Concentration.....	3.3.1-1
Table 3-1	MPC Cavity Drying Limits.....	3.4-1
Table 3-2	MPC Helium Backfill Limits .....	3.4-2
4.0	.....	4.0-1
5.0	ADMINISTRATIVE CONTROLS.....	5.0-1
5.1	Deleted	
5.2	Deleted	
5.3	Deleted	
5.4	Radioactive Effluent Control Program .....	5.0-1
5.5	Cask Transport Evaluation Program .....	5.0-2
5.6	Deleted	
5.7	Radiation Protection Program.....	5.0-5
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements .....	5.0-6

---

## 1.0 USE AND APPLICATION

### 1.1 Definitions

---

-----NOTE-----

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

-----

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
FUEL BUILDING	The FUEL BUILDING is the site-specific power plant facility, governed by the regulations of 10CFR Part 50, where the loaded OVERPACK or TRANSFER CASK is transferred to or from the transporter.
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 included MPC transfer between the TRANSFER CASK and the OVERPACK.
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.

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(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 OVERPACK does not include the TRANSFER CASK.
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 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 sitting on a storage pad within the ISFSI perimeter. STORAGE OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
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. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.

---

(continued)

## 1.1 Definitions (continued)

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### TRANSPORT OPERATIONS

TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS includes transfer of the MPC between the OVERPACK and the TRANSFER CASK.

### UNLOADING OPERATIONS

UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.

---

## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

---

**PURPOSE**            The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

---

**BACKGROUND**        Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

---

(continued)

1.2 Logical Connectors

---

EXAMPLES      The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . .  <u>AND</u>  A.2 Restore . . .	

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

(continued)

---



1.2 Logical Connectors

EXAMPLES  
(continued)

EXAMPLE 1.2-2

ACTIONS

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

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

OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

## 1.0 USE AND APPLICATION

### 1.3 Completion Times

---

**PURPOSE** The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

---

**BACKGROUND** Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

---

**DESCRIPTION** The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the HI-STORM 100 System is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the HI-STORM 100 System is not within the LCO Applicability.

Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

---

(continued)

1.3 Completion Times (continued)

EXAMPLES

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

EXAMPLE 1.3-1

ACTIONS

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-3

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times (continued)

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

### 1.4 Frequency

---

**PURPOSE**            The purpose of this section is to define the proper use and application of Frequency requirements.

---

**DESCRIPTION**      Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

---

(continued)

1.4 Frequency (continued)

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EXAMPLES

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

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

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

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

(continued)

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1.4 Frequency

---

EXAMPLES  
(continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

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

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

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

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

---

2.0

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This section is intentionally left blank

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**3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY**

---

LCO 3.0.1            LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.

---

LCO 3.0.2            Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

---

LCO 3.0.3            Not applicable.

---

LCO 3.0.4            When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.

---

LCO 3.0.5            Equipment removed from service or not in service in compliance with ACTIONS maybe returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

---

### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

SR 3.0.1           SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.

---

SR 3.0.2           The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply. If a Completion Time requires periodic performance on a “once per...” basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

---

SR 3.0.3           If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

#### SR 3.0.3

(continued)

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

---

#### SR 3.0.4

Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

---



ACTIONS  
(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Actions and associated Completion Times not met.	C.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify that the MPC cavity has been dried in accordance with the applicable limits in Table 3-1.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill quantity is within the limit specified in Table 3-2 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be operable

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System inoperable.	A.1 Restore SFSC Heat Removal System to operable status.	8 hours
B. Required Action A.1 and associated Completion Time not met.	B.1 Measure SFSC dose rates in accordance with the Radiation Protection Program.	Immediately and once per 12 hours thereafter
	<u>AND</u>	
	B.2.1 Restore SFSC Heat Removal System to operable status.	64 hours
	<u>OR</u>	
	B.2.2 Transfer the MPC into a TRANSFER CASK.	64 hours



SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify all OVERPACK inlet and outlet air ducts are free of blockage.	24 hours
	<u>OR</u>	
	For OVERPACKS with installed temperature monitoring equipment, verify that the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is $\leq 126^{\circ}\text{F}$ .	24 hours

3.1 SFSC INTEGRITY

3.1.3 Fuel Cool-Down

LCO 3.1.3            The MPC cavity bulk helium temperature shall be  $\leq 200^\circ$  F

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

APPLICABILITY:    UNLOADING OPERATIONS prior to re-flooding.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity bulk helium temperature not within limit.	A.1 Establish MPC cavity bulk helium temperature within limit.	Prior to initiating MPC re-flooding operations
	<u>AND</u> A.2 Ensure adequate heat transfer from the MPC to the environment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1    Ensure via analysis or direct measurement of MPC exit gas temperature that MPC cavity bulk helium temperature is within limit.	Prior to MPC re-flooding operations.

3.1 SFSC INTEGRITY

3.1.4 Supplemental Cooling System

LCO 3.1.4 The Supplemental Cooling System (SCS) shall be operable

-----NOTE-----  
Upon reaching steady state operation, the SCS may be temporarily disabled for a short duration ( $\leq 7$  hours) to facilitate necessary operational evolutions, such as movement of the TRANSFER CASK through a door way, or other similar operation.  
-----

APPLICABILITY: This LCO is applicable when the loaded MPC is in the TRANSFER CASK and:

- a. Within 4 hours of the completion of MPC drying operations in accordance with LCO 3.1.1 or within 4 hours of transferring the MPC into the TRANSFER CASK if the MPC is to be unloaded

AND

- b. The MPC contains one or more fuel assemblies with an average burnup  $> 45,000$  MWD/MTU

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Supplemental Cooling System inoperable.	A.1 Restore SFSC Supplemental Cooling System to operable status.	7 days
B. Required Action A.1 and associated Completion Time not met.	B.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify Supplemental Cooling System is operable.	2 hours

Deleted |  
3.2.1

3.2 Deleted. |

3.2.1 Deleted. |

LCO 3.2.1 Deleted. |  
|

3.2 SFSC RADIATION PROTECTION

3.2.2 TRANSFER CASK Surface Contamination

LCO 3.2.2            Removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC shall each not exceed:

- a. 1000 dpm/100 cm<sup>2</sup> from beta and gamma sources
- b. 20 dpm/100 cm<sup>2</sup> from alpha sources.

-----NOTE-----

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

-----  
--

APPLICABILITY:    During TRANSPORT OPERATIONS.

ACTIONS

-----NOTE-----

-  
Separate Condition entry is allowed for each TRANSFER CASK.

-----  
--

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify that the removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC containing fuel is within limits.	Once, prior to TRANSPORT OPERATIONS

Deleted |  
3.2.3

3.2 Deleted. |

3.2.3 Deleted. |

LCO 3.2.3 Deleted. |

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%  $^{235}\text{U}$ :  $\geq 400$  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%  $^{235}\text{U}$ :  $\geq 300$  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%  $^{235}\text{U}$  and  $\leq 5.0$  wt%  $^{235}\text{U}$ :  $\geq 600$  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	
	Initial Enrichment $\leq 4.1$ wt% $^{235}\text{U}$ (ppmb)	Initial Enrichment $> 4.1$ wt% and $\leq 5.0$ wt% $^{235}\text{U}$ (ppmb)	Initial Enrichment $\leq 4.1$ wt% $^{235}\text{U}$ (ppmb)	Initial Enrichment $> 4.1$ wt% and $\leq 5.0$ wt% $^{235}\text{U}$ (ppmb)
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A	1,300	1,900	1,500	2,300
17x17A/B/C	1,900	2,600	2,100	2,900



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

AND

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

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- This surveillance is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. -----	Once, within 4 hours prior to entering the Applicability of this LCO.
SR 3.3.1.1      Verify boron concentration is within the applicable limit using two independent measurements.	<u>AND</u> Once per 48 hours thereafter.

Table 3-1  
MPC Cavity Drying Limits

<b>Fuel Burnup (MWD/MTU)</b>	<b>MPC Heat Load (kW)</b>	<b>Method of Moisture Removal (Notes 1 and 2)</b>
All Assemblies $\leq$ 45,000	$\leq$ 28.74	VDS or FHD
One or more assemblies $>$ 45,000	$\leq$ 28.74	FHD

Notes:

1. VDS means Vacuum Drying System. The acceptance criterion for VDS is MPC cavity pressure shall be  $\leq$  3 torr for  $\geq$  30 minutes.
2. FHD means Forced Helium Dehydration System. The acceptance criterion for the FHD System is gas temperature exiting the demoinsturizer shall be  $\leq$  21°F for  $\geq$  30 minutes or gas dew point exiting the MPC shall be  $\leq$  22.9°F for  $\geq$  30 minutes .

Table 3-2  
MPC Helium Backfill Limits<sup>1</sup>

MPC MODEL	LIMIT
MPC-24/24E/24EF	
i. Cask Heat Load $\leq$ 27.77 kW (MPC-24) or $\leq$ 28.17 kW (MPC-24E/EF)	0.1212 +/-10% g-moles/l  OR  $\geq$ 29.3 psig and $\leq$ 33.3 psig
MPC-68/68F/68FF	
i. Cask Heat Load $\leq$ 28.19 kW	0.1218 +/-10% g-moles/l  OR  $\geq$ 29.3 psig and $\leq$ 33.3 psig
MPC-32/32F	
Cask Heat Load $\leq$ 28.74 kW	$\geq$ 29.3 psig and $\leq$ 33.3 psig

<sup>1</sup> Helium used for backfill of MPC shall have a purity of  $\geq$  99.995%. Pressure range is at a reference temperature of 70°F

4.0

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## 5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

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The following programs shall be established, implemented and maintained.

5.1 Deleted.

5.2 Deleted.

5.3 Deleted.

5.4 Radioactive Effluent Control Program

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

- a. The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.
- b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.
- c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

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(continued)

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 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.).

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, 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. For site-specific transport route surfaces that are not bounded by either the Set A or Set B parameters of FSAR Table 2.2.9, 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)

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## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 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 transportation between the FUEL BUILDING and the CTF and/or ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 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 transportation between the FUEL BUILDING and the CTF and/or ISFSI pad provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.

(continued)

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ADMINISTRATIVE CONTROLS AND PROGRAMS

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5.5 Cask Transport Evaluation Program (continued)

Table 5-1

TRANSFER CASK and Free-Standing OVERPACK Lifting Requirements

ITEM	ORIENTATION	LIFTING HEIGHT LIMIT (in.)
TRANSFER CASK	Horizontal	42 (Notes 1 and 2)
TRANSFER CASK	Vertical	None Established (Note 2)
OVERPACK	Horizontal	Not Permitted
OVERPACK	Vertical	11 (Note 3)

- Notes:
1. To be measured from the lowest point on the TRANSFER CASK (i.e., the bottom edge of the cask/lid assemblage)
  2. See Technical Specification 5.5.a.3 and 4
  3. See Technical Specification 5.5.a.3.

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(continued)



## ADMINISTRATIVE CONTROLS AND PROGRAMS

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5.6 Deleted.

### 5.7 Radiation Protection Program

- 5.7.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for measuring dose rates, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). This program provides a means to help ensure that licensees using the HI-STORM 100 System do not violate the dose limits in 10 CFR 72. The actions and criteria to be included in the program are provided below.
- 5.7.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.
- 5.7.3 Based on the analysis performed pursuant to Section 5.7.2, the licensee shall establish cask surface dose rate limits for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK to be used at the site. Average neutron and gamma dose rate limits shall be established at the following locations:
- a. The top of the TRANSFER CASK and the OVERPACK.
  - b. The side of the TRANSFER CASK and OVERPACK
  - c. The average of the inlet and outlet ducts on the OVERPACK
- 5.7.4 Notwithstanding the limits established in Section 5.7.3, the measured dose rates on a loaded OVERPACK shall not exceed the following values:
- a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK
  - b. 125 mrem/hr (gamma + neutron) on the side of the OVERPACK
  - c. 130 mrem/hr (gamma + neutron) at the inlet and outlet vent ducts
- 5.7.5 The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates and calculate average values as described in Section 5.7.8 and 5.7.9 for comparison against the limits established in Section 5.7.3 or Section 5.7.4, whichever are lower.

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 5.7 Radiation Protection Program (cont'd)

- 5.7.6 If the measured average surface dose rates exceed the lower of the two limits established in Section 5.7.3 or Section 5.7.4, the licensee shall:
- a. Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.
  - b. Perform an evaluation to verify whether placement of the as-loaded OVERPACK at the ISFSI will cause the dose limits of 10 CFR 72.104 to be exceeded.
- 5.7.7 If the evaluation performed pursuant to Section 5.7.6 shows that the dose limits of 10 CFR 72.104 will be exceeded, appropriate corrective action shall be taken to ensure the dose limits are not exceeded.
- 5.7.8 TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:
- a. A minimum of 12 dose rate measurements shall be taken on the side of the TRANSFER CASK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively. Within each set, the measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.
  - b. A minimum of four (4) TRANSFER CASK top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid.
  - c. A minimum of 12 dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.
  - d. A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top shield, 90 degrees apart around the circumference of the lid.

## ADMINISTRATIVE CONTROLS AND PROGRAMS

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### 5.7 Radiation Protection Program (cont'd)

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

5.7.9 The average measured dose rates shall be calculated by summing the individual neutron and gamma dose rates measured in Sections 5.7.8.a through 5.7.8.e and dividing by the total number of measurements for that section. The neutron and gamma dose rates shall be averaged separately.

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**CERTIFICATE OF COMPLIANCE NO. 1014**

**APPENDIX B**

**APPROVED CONTENTS AND DESIGN FEATURES**

**FOR THE HI-STORM 100 CASK SYSTEM**

## 1.0 Definitions

## -----NOTE-----

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

<u>Term</u>	<u>Definition</u>
CASK TRANSFER FACILITY (CTF)	The CASK TRANSFER FACILITY includes the following components and equipment: (1) a Cask Transfer Structure used to stabilize the TRANSFER CASK and MPC during lifts involving spent fuel not bounded by the regulations of 10 CFR Part 50, and (2) Either a stationary lifting device or a mobile lifting device used in concert with the stationary structure to lift the OVERPACK, TRANSFER CASK, and MPC
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STORM 100 System are as follows: <ol style="list-style-type: none"> <li>1. Holtec Dresden Unit 1/Humboldt Bay design</li> <li>2. Transnuclear Dresden Unit 1 design</li> <li>3. Holtec Generic BWR design</li> <li>4. Holtec Generic PWR design</li> </ol>
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

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(continued)

1.0 Definitions (continued)

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INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).
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 included MPC transfer between the TRANSFER CASK and the OVERPACK.
MINIMUM ENRICHMENT	MINIMUM ENRICHMENT is the minimum assembly average enrichment. Natural uranium 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.
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), water displacement guide tube plugs, orifice rod assemblies, and vibration suppressor inserts.
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 OVERPACK does not include the TRANSFER CASK.

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(continued)

1.0 Definitions (continued)

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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.
SPENT FUEL STORAGE CASKS (SFSCs)	An SFSC is a container approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK and its integral MPC.
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. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OPERATIONS include transfer of the MPC between the OVERPACK and the TRANSFER CASK.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.
ZR	ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor.

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## 2.0 APPROVED CONTENTS

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### 2.1 Fuel Specifications and Loading Conditions

#### 2.1.1 Fuel To Be Stored In The HI-STORM 100 SFSC System

- a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM 100 SFSC System.
- b. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the decay heat generation limit for the stainless steel clad fuel assemblies.
- c. For MPCs partially loaded with DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, all remaining ZR clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the DAMAGED FUEL ASSEMBLIES. This requirement applies only to uniform fuel loading.
  - a. For MPCs partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining ZR clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the 6x6A, 6x6B, 6x6C, 7x7A and 8x8A fuel assemblies.
  - b. All BWR fuel assemblies may be stored with or without ZR channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without ZR or stainless steel channels.

#### 2.1.2 Uniform Fuel Loading

Any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions related to DAMAGED FUEL, FUEL DEBRIS, and NON-FUEL HARDWARE specified in the CoC.

(continued)

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## 2.0 Approved Contents

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### 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 those 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, and MPC-68FF models, respectively<sup>1</sup>. 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 Deviations from Cask Contents Requirements

Proposed alternatives to the contents listed in Section 2.0 may be authorized on a case-specific basis by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative contents should demonstrate that:

2.3.1 The proposed alternative contents would provide an equivalent level of safety, and

2.3.2 The proposed alternative contents are consistent with the applicable requirements.

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<sup>1</sup> 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.

## 2.0 APPROVED CONTENTS

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### 2.3 Deviations from Cask Contents Requirements (cont'd)

2.3.3 Requests for case-specific NRC approval of alternatives to contents shall be submitted in accordance with 10 CFR 72.4 by the certificate holder. Case-specific alternatives approved pursuant to this section shall be incorporated permanently into the CoC by the certificate holder in accordance with 10 CFR 72.244. Requests made pursuant to this section must meet all of the following requirements:

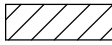
2.3.3.1 The proposed change must not significantly decrease any safety margins as described in the HI-STORM 100 System FSAR, as updated.


2.3.3.2 The proposed change may involve only the physical fuel assembly parameters listed below as specified in Tables 2.1-1, 2.1-2, and/or 2.1-3 of this Appendix:

- a. Fuel Assembly Length
- b. Fuel Assembly Width
- c. Fuel Assembly Weight
- d. Fuel Rod Clad Outside Diameter (OD)
- e. Fuel Rod Clad Inside Diameter (ID)
- f. Fuel Pellet Diameter
- g. Fuel Rod Pitch
- h. PWR Guide/Instrument Tube Thickness
- i. BWR Water Rod Thickness
- j. BWR Channel Thickness

2.3.3.3 The proposed change must be required to meet a compelling user need whereby using the normal certificate amendment process is not practical.

LEGEND:

REGION 1: 

REGION 2: 

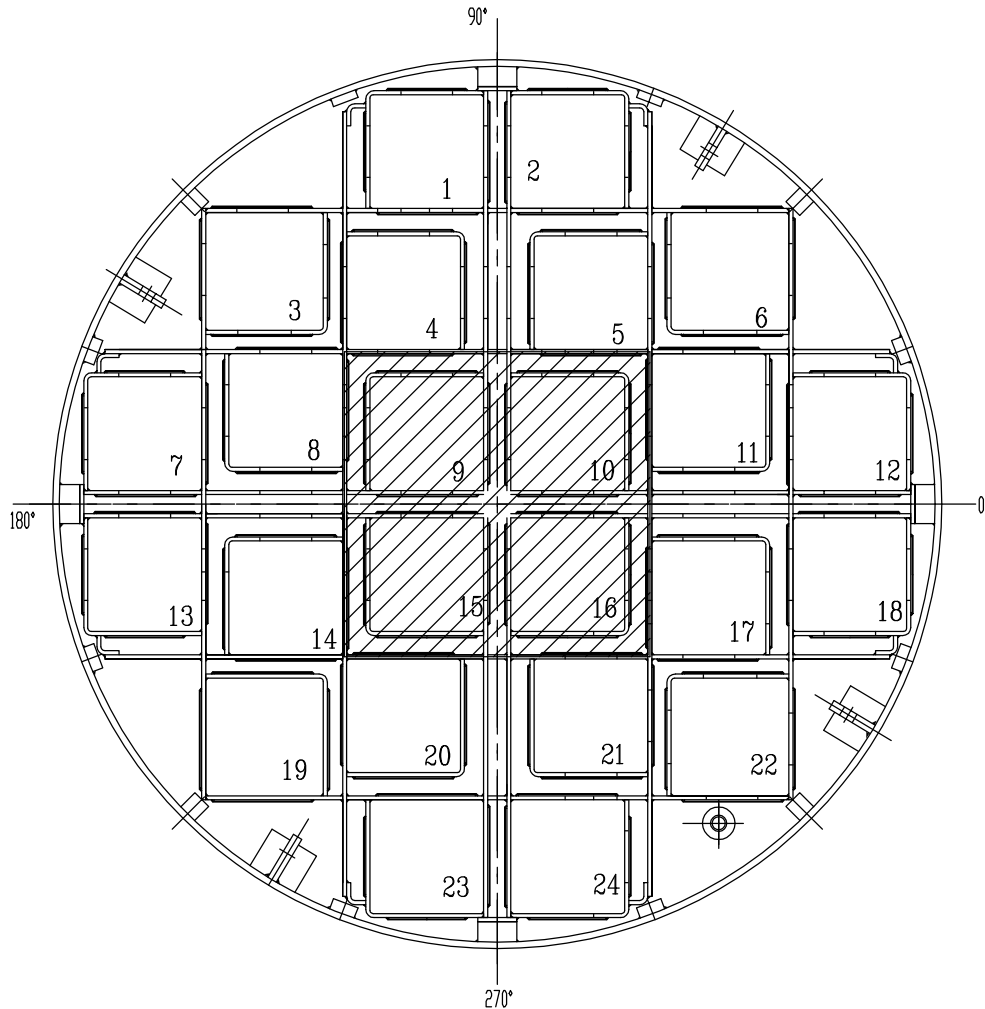
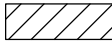



FIGURE 2.1-1  
FUEL LOADING REGIONS - MPC-24

LEGEND:

REGION 1: 

REGION 2: 

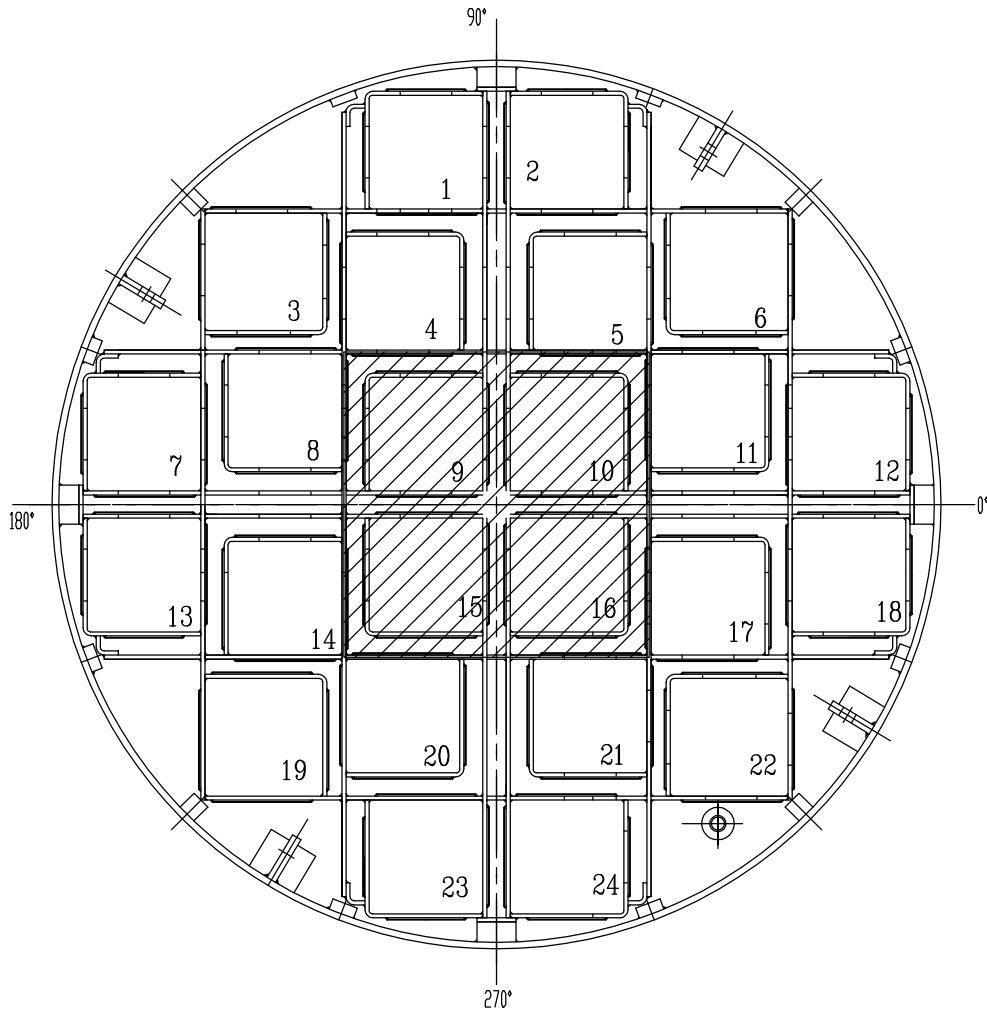

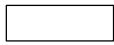


FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: 

REGION 2: 

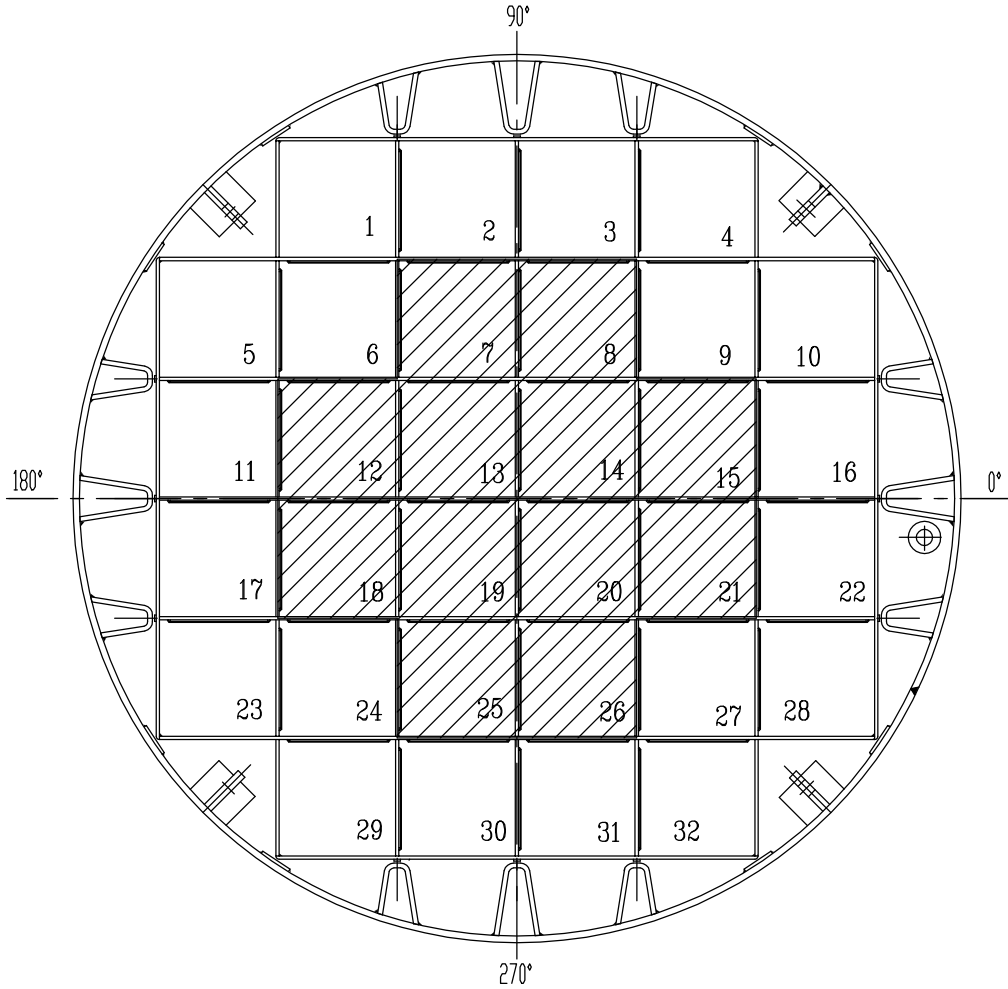




FIGURE 2.1-3

FUEL LOADING REGIONS – MPC-32/32F

LEGEND:

REGION 1: 

REGION 2: 

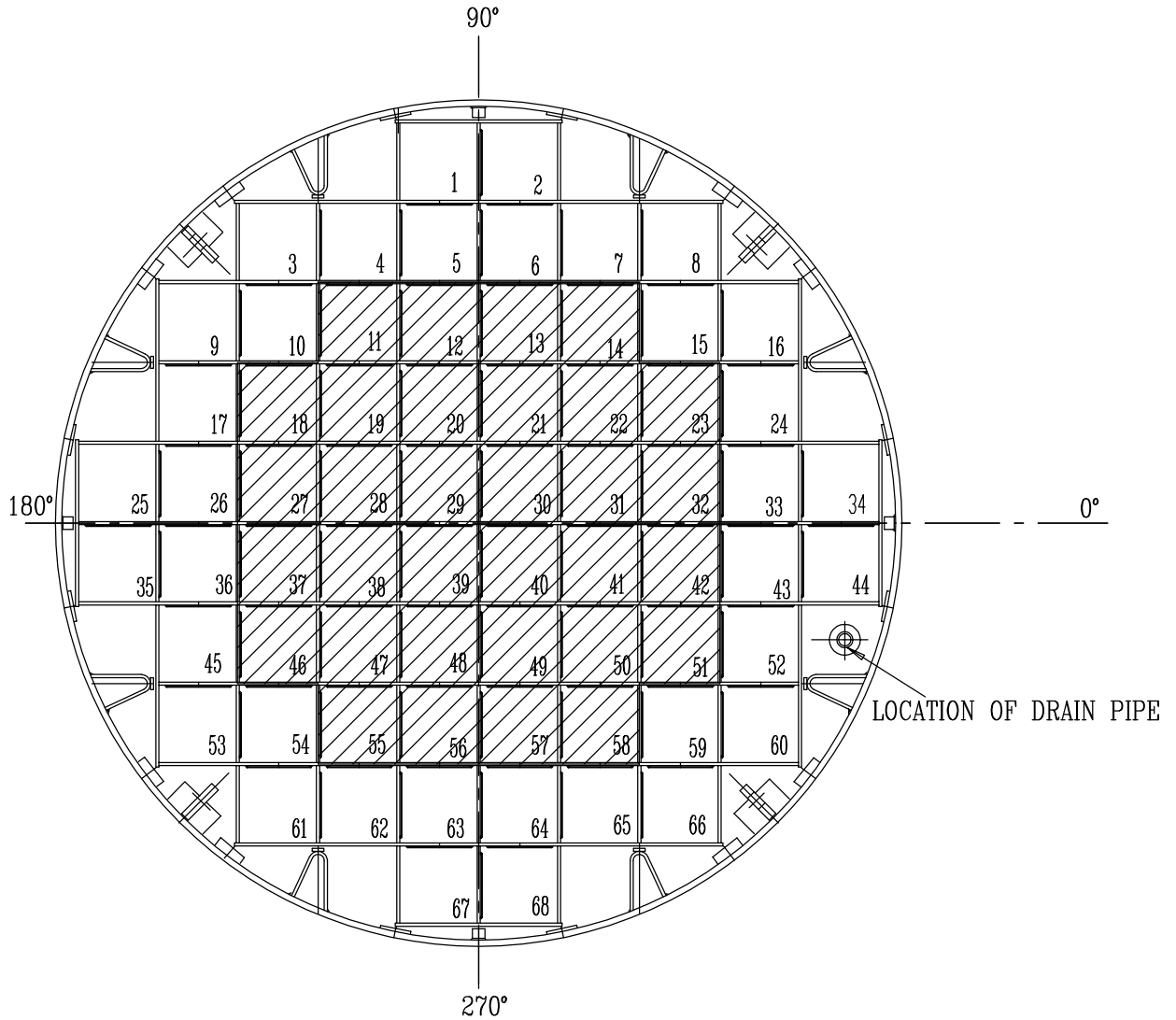


FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of 39)  
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  $\geq$  8 years and an average burnup  $\leq$  40,000 MWD/MTU.

ii. All Other Array/Classes Cooling time and average burnup as specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 2 of 39)  
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq 710$  Watts

ii All Other Array/Classes As specified in Section 2.4.

e. Fuel Assembly Length:  $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width:  $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight:  $\leq 1,680$  lbs (including NON-FUEL HARDWARE)

B. Quantity per MPC: Up to 24 fuel assemblies.

C. Deleted.

D. Neutron sources and DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading into the MPC-24.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.



Table 2.1-1 (page 3 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, 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, 6x6C, 7x7A, and 8x8A:                      | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU                             |
| ii. Array/Class 8x8F  | Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.                            |
| iii. Array/Classes 10x10D and 10x10E                              | Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.                            |
| iv. All Other Array/Classes                                       | As specified in Section 2.4.   |

Table 2.1-1 (page 4 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

- |  |                              |
|--|------------------------------|
| i. Array/Classes 6x6A, 6x6C,<br>7x7A, and 8x8A | $\leq 115$ Watts             |
| ii. Array/Class 8x8F                           | $\leq 183.5$ Watts.          |
| iii. Array/Classes 10x10D and<br>10x10E        | $\leq 95$ Watts              |
| iv. All Other Array/Classes                    | As specified in Section 2.4. |

- |                          |                                      |
|--------------------------|--------------------------------------|
| f. Fuel Assembly Length: | $\leq 176.5$ inches (nominal design) |
| g. Fuel Assembly Width:  | $\leq 5.85$ inches (nominal design)  |
| h. Fuel Assembly Weight: | $\leq 700$ lbs, including channels   |

Table 2.1-1 (page 5 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 and meet 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:                     |  |
| i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A                       | As specified in Table 2.1-3 for the applicable fuel assembly array/class.                            |
| ii. All Other Array/Classes specified in Table 2.1-3              | 4.0 wt% <sup>235</sup> 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/Classes 6x6A, 6x6C, 7x7A, and 8x8A                       | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.                            |
| ii. Array/Class 8x8F  | Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.                            |
| iii. Array/Classes 10x10D and 10x10E                              | Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.                            |
| iv. All Other Array Classes                                       | As specified in Section 2.4.   |

Table 2.1-1 (page 6 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

- |      |   |                              |
|------|---|------------------------------|
| i.   | Array/Class 6x6A, 6x6C, 7x7A,<br>and 8x8A | $\leq 115$ Watts             |
| ii.  | Array/Class 8x8F                          | $\leq 183.5$ Watts           |
| iii. | Array/Classes 10x10D and<br>10x10E        | $\leq 95$ Watts              |
| iv.  | All Other Array/Classes                   | As specified in Section 2.4. |

f. Fuel Assembly Length:

- |     |  |                                      |
|-----|--|--------------------------------------|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq 135.0$ inches (nominal design) |
| ii. | All Other Array/Classes                  | $\leq 176.5$ inches (nominal design) |

g. Fuel Assembly Width:

- |     |  |                                     |
|-----|--|-------------------------------------|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq 4.70$ inches (nominal design) |
| ii. | All Other Array/Classes                  | $\leq 5.85$ inches (nominal design) |

h. Fuel Assembly Weight:

- |     |  |  |
|-----|--|--|
| i.  | Array/Class 6x6A, 6x6C, 7x7A,<br>or 8x8A | $\leq 550$ lbs, including channels and DFC |
| ii. | All Other Array/Classes                  | $\leq 700$ lbs, including channels and DFC |

Table 2.1-1 (page 7 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 8 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for array/class 6x6B.                           |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for array/class 6x6B.                           |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                  |

Table 2.1-1 (page 9 of 39)  
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods ( $\text{ThO}_2$  and  $\text{UO}_2$ ) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | ZR   |
| b. Composition:   | 98.2 wt.% $\text{ThO}_2$ , 1.8 wt. % $\text{UO}_2$ with an enrichment of 93.5 wt. % $^{235}\text{U}$ . |
| c. Number of Rods Per Thoria Rod Canister:  | $\leq 18$  |
| d. Decay Heat Per Thoria Rod Canister:  | $\leq 115$ Watts   |
| e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister: | A fuel post-irradiation cooling time $\geq 18$ years and an average burnup $\leq 16,000$ MWD/MTIHM.    |
| f. Initial Heavy Metal Weight:  | $\leq 27$ kg/canister  |
| g. Fuel Cladding O.D.:  | $\geq 0.412$ inches  |
| h. Fuel Cladding I.D.:  | $\leq 0.362$ inches  |
| i. Fuel Pellet O.D.:  | $\leq 0.358$ inches  |
| j. Active Fuel Length:  | $\leq 111$ inches  |
| k. Canister Weight:   | $\leq 550$ lbs, including fuel   |

Table 2.1-1 (page 10 of 39)  
Fuel Assembly Limits

---

II. MPC MODEL: MPC-68 (continued)

B. Quantity per MPC:

1. Up to one (1) Dresden Unit 1 Thoria Rod Canister;
  2. Up to 68 array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS;
  3. Up to sixteen (16) other BWR DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68; and/or
  4. Any number of BWR INTACT FUEL ASSEMBLIES up to a total of 68.
- C. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.
- D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.
- E. FUEL DEBRIS is not authorized for loading in the MPC-68.



Table 2.1-1 (page 11 of 39)  
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without ZR channels. Uranium oxide BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array class 6x6A, 6x6C, 7x7A or 8x8A, 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: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)                                      |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)                                       |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 12 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without ZR channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, 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: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU. |
| e. Decay Heat Per Assembly:                                       | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)                                      |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)                                       |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                |

Table 2.1-1 (page 13 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without ZR channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the uranium oxide BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, 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 original fuel assembly array/class.                       |
| c. Initial Maximum Rod Enrichment:                               | As specified in Table 2.1-3 for the applicable original fuel assembly array/class.                       |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU for the original fuel assembly. |
| e. Decay Heat Per Assembly                                       | $\leq$ 115 Watts   |
| f. Original Fuel Assembly Length                                 | $\leq$ 135.0 inches (nominal design)   |
| g. Original Fuel Assembly Width                                  | $\leq$ 4.70 inches (nominal design)  |
| h. Fuel Debris Weight  | $\leq$ 550 lbs, including channels and DFC   |

Table 2.1-1 (page 14 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without ZR channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 400 lbs, including channels  |

Table 2.1-1 (page 15 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without ZR channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |   |
|---|---|
| a. Cladding Type:   | ZR  |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for fuel assembly array/class 6x6B.             |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts  |
| f. Fuel Assembly Length:  | $\leq$ 135.0 inches (nominal design)  |
| g. Fuel Assembly Width:   | $\leq$ 4.70 inches (nominal design)   |
| h. Fuel Assembly Weight:  | $\leq$ 550 lbs, including channels and DFC                                  |

Table 2.1-1 (page 16 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without ZR channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the MOX BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | ZR   |
| b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:                     | As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.                                   |
| c. Initial Maximum Rod Enrichment:                                | As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.                                   |
| d. Post-irradiation Cooling Time and Average Burnup Per Assembly: | Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM for the original fuel assembly. |
| e. Decay Heat Per Assembly  | $\leq$ 115 Watts   |
| f. Original Fuel Assembly Length:                                 | $\leq$ 135.0 inches (nominal design)   |
| g. Original Fuel Assembly Width:                                  | $\leq$ 4.70 inches (nominal design)  |
| h. Fuel Debris Weight:  | $\leq$ 550 lbs, including channels and DFC   |

Table 2.1-1 (page 17 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

7. Thoria rods ( $\text{ThO}_2$  and  $\text{UO}_2$ ) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

- |   |  |
|---|--|
| a. Cladding Type:   | ZR   |
| b. Composition:   | 98.2 wt.% $\text{ThO}_2$ , 1.8 wt. % $\text{UO}_2$ with an enrichment of 93.5 wt. % $^{235}\text{U}$ . |
| c. Number of Rods Per Thoria Rod Canister:  | $\leq 18$  |
| d. Decay Heat Per Thoria Rod Canister:  | $\leq 115$ Watts   |
| e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister: | A fuel post-irradiation cooling time $\geq 18$ years and an average burnup $\leq 16,000$ MWD/MTIHM.    |
| f. Initial Heavy Metal Weight:  | $\leq 27$ kg/canister  |
| g. Fuel Cladding O.D.:  | $\geq 0.412$ inches  |
| h. Fuel Cladding I.D.:  | $\leq 0.362$ inches  |
| i. Fuel Pellet O.D.:  | $\leq 0.358$ inches  |
| j. Active Fuel Length:  | $\leq 111$ inches  |
| k. Canister Weight:   | $\leq 550$ lbs, including fuel   |

Table 2.1-1 (page 18 of 39)  
Fuel Assembly Limits

---

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

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

(All fuel assemblies must be array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A):

Up to four (4) DFCs containing uranium oxide BWR FUEL DEBRIS or MOX BWR FUEL DEBRIS. The remaining MPC-68F fuel storage locations may be filled with fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES;
2. MOX BWR INTACT FUEL ASSEMBLIES;
3. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES placed in DFCs;
4. MOX BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium source material shall be in a water rod location.



Table 2.1-1 (page 19 of 39)  
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.                            |
| ii. All Other Array/Classes                                       | As specified in Section 2.4.  |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.  |

Table 2.1-1 (page 20 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

- |   |  |   |  |
|---|--|---|--|
| d. Decay Heat Per Fuel Storage Location:    |  |   |  |
| i. Array/Classes 14x14D, 14x14E, and 15x15G |  | ≤ 710 Watts.                              |  |
| ii. All other Array/Classes                 |  | As specified in Section 2.4.              |  |
| e. Fuel Assembly Length:                    |  | ≤ 176.8 inches (nominal design)           |  |
| f. Fuel Assembly Width:                     |  | ≤ 8.54 inches (nominal design)            |  |
| g. Fuel Assembly Weight:                    |  | ≤ 1,680 lbs (including NON-FUEL HARDWARE) |  |

Table 2.1-1 (page 21 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES 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  $\geq$  8 years and an average burnup  $\leq$  40,000 MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 22 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

- |   |   |
|---|---|
| d. Decay Heat Per Fuel Storage Location:    |   |
| i. Array/Classes 14x14D, 14x14E, and 15x15G | ≤ 710 Watts.                                      |
| ii. All Other Array/Classes                 | As specified in Section 2.4.                      |
| e. Fuel Assembly Length                     | ≤ 176.8 inches (nominal design)                   |
| f. Fuel Assembly Width                      | ≤ 8.54 inches (nominal design)                    |
| g. Fuel Assembly Weight                     | ≤ 1,680 lbs (including NON-FUEL HARDWARE and DFC) |

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron sources and FUEL DEBRIS are not authorized for loading in the MPC-24E.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration supressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 23 of 39)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32

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  $\geq$  9 years and an average burnup  $\leq$  30,000 MWD/MTU or cooling time  $\geq$  20 years and an average burnup  $\leq$  40,000 MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 24 of 39)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage  
Location:

≤ 500 Watts

i. Array/Classes 14x14D,  
14x14E, and 15x15G

As specified in Section 2.4.

ii. All Other Array/Classes

e. Fuel Assembly Length

≤ 176.8 inches (nominal design)

f. Fuel Assembly Width

≤ 8.54 inches (nominal design)

g. Fuel Assembly Weight

≤ 1,680 lbs (including NON-FUEL  
HARDWARE)

Table 2.1-1 (page 25 of 39)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES 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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU. |
| ii. All Other Array/Classes                                       | As specified in Section 2.4.   |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.   |

Table 2.1-1 (page 26 of 39)  
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel  
Storage Location:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq$  500 Watts.

ii. All Other Array/Classes As specified in Section 2.4.

e. Fuel Assembly Length  $\leq$  176.8 inches (nominal design)

f. Fuel Assembly Width  $\leq$  8.54 inches (nominal design)

g. Fuel Assembly Weight  $\leq$  1,680 lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to eight (8) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32 fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron sources and FUEL DEBRIS are not authorized for loading in the MPC-32.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 13, 14, 19, and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.



Table 2.1-1 (page 27 of 39)  
Fuel Assembly Limits

VI. MPC MODEL: 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 $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU (or MTU/MTIHM).            |
| ii. Array/Class 8x8F   | Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.                           |
| iii. Array/Classes 10x10D and 10x10E                             | Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.                           |
| iv. All Other Array/Classes                                      | As specified in Section 2.4.  |

Table 2.1-1 (page 28 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

- |  |                              |
|--|------------------------------|
| i. Array/Classes 6x6A, 6X6b,<br>6x6C, 7x7A, and 8x8A | ≤ 115 Watts                  |
| ii. Array/Class 8x8F                                 | ≤ 183.5 Watts                |
| iii. Array/Classes 10x10D and<br>10x10E              | ≤ 95 Watts                   |
| iv. All Other Array/Classes                          | As specified in Section 2.4. |

f. Fuel Assembly Length

- |   |                                 |
|---|---------------------------------|
| i. Array/Class 6x6A, 6x6B,<br>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,<br>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,<br>7x7A, or 8x8A | ≤ 550 lbs, including channels |
| ii. All Other Array/Classes                       | ≤ 700 lbs, including channels |

Table 2.1-1 (page 29 of 39)  
Fuel Assembly Limits

VI. MPC MODEL: 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                                       | $\leq 4.0$ wt.% <sup>235</sup> 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 $\geq 18$ years and an average burnup $\leq 30,000$ MWD/MTU (or MWD/MTIHM).                |
| ii. Array/Class 8x8F  | Cooling time $\geq 10$ years and an average burnup $\leq 27,500$ MWD/MTU.                               |
| iii. Array/Class 10x10D and 10x10E                                | Cooling time $\geq 10$ years and an average burnup $\leq 22,500$ MWD/MTU.                               |
| iv. All Other Array/Classes                                       | As specified in Section 2.4.  |

Table 2.1-1 (page 30 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

- |      |  |                              |  |
|------|--|------------------------------|--|
| i.   | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 115 Watts             |  |
| ii.  | Array/Class 8x8F                               | $\leq$ 183.5 Watts           |  |
| iii. | Array/Classes 10x10D and<br>10x10E             | $\leq$ 95 Watts              |  |
| iv.  | All Other Array/Classes                        | As specified in Section 2.4. |  |

f. Fuel Assembly Length

- |     |  |                                      |  |
|-----|--|--------------------------------------|--|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 135.0 inches (nominal design) |  |
| ii. | All Other Array/Classes                        | $\leq$ 176.5 inches (nominal design) |  |

g. Fuel Assembly Width

- |     |  |                                     |  |
|-----|--|-------------------------------------|--|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 4.70 inches (nominal design) |  |
| ii. | All Other Array/Classes                        | $\leq$ 5.85 inches (nominal design) |  |

h. Fuel Assembly Weight

- |     |  |  |  |
|-----|--|--|--|
| i.  | Array/Class 6x6A, 6x6B, 6x6C,<br>7x7A, or 8x8A | $\leq$ 550 lbs, including channels and DFC |  |
| ii. | All Other Array/Classes                        | $\leq$ 700 lbs, including channels and DFC |  |

Table 2.1-1 (page 31 of 39)  
Fuel Assembly limits

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

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

1. For fuel assembly array/classes 6x6A, 6X6B, 6x6C, 7x7A, or 8x8A, up to 68 BWR INTACT FUEL ASSEMBLIES and/or DAMAGED FUEL ASSEMBLIES. Up to eight (8) DFCs containing FUEL DEBRIS from these array/classes may be stored.
2. For all other array/classes, 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 MPC-68FF fuel storage locations may be filled with fuel assemblies of the following type:
  - i. Uranium Oxide BWR INTACT FUEL ASSEMBLIES; or
  - ii. MOX BWR INTACT FUEL ASSEMBLIES.

C. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68FF. The Antimony-Beryllium source material shall be in a water rod location.

D. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

Table 2.1-1 (page 32 of 39)  
Fuel Assembly Limits

VII. MPC MODEL: 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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.                            |
| ii. All Other Array/Classes                                       | As specified in Section 2.4.  |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.  |

Table 2.1-1 (page 33 of 39)  
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

- |   |   |  |
|---|---|--|
| d. Decay Heat Per Fuel Storage Location:    |   |  |
| i. Array/Classes 14x14D, 14x14E, and 15x15G | ≤ 710 Watts.                              |  |
| ii. All other Array/Classes                 | As specified in Section 2.4.              |  |
| e. Fuel Assembly Length:                    | ≤ 176.8 inches (nominal design)           |  |
| f. Fuel Assembly Width:                     | ≤ 8.54 inches (nominal design)            |  |
| g. Fuel Assembly Weight:                    | ≤ 1,680 lbs (including NON-FUEL HARDWARE) |  |

Table 2.1-1 (page 34 of 39)  
Fuel Assembly Limits

VII. MPC MODEL: 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 $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.                            |
| ii. All Other Array/Classes                                       | As specified in Section 2.4.  |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.  |



Table 2.1-1 (page 35 of 39)  
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

- |  |   |
|--|---|
| d. Decay Heat Per Fuel<br>Storage Location:    | $\leq 710$ Watts.   |
| i. Array/Classes 14x14D,<br>14x14E, and 15x15G | As specified in Section 2.4.                              |
| ii. All Other Array/Classes                    |   |
| e. Fuel Assembly Length                        | $\leq 176.8$ inches (nominal design)                      |
| f. Fuel Assembly Width                         | $\leq 8.54$ inches (nominal design)                       |
| g. Fuel Assembly Weight                        | $\leq 1,680$ lbs (including NON-FUEL<br>HARDWARE and DFC) |

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24EF fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron sources are not permitted for loading in the MPC-24EF.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 36 of 39)  
Fuel Assembly Limits

---

VIII. MPC MODEL: 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  $\geq 9$  years and an average burnup  $\leq 30,000$  MWD/MTU or cooling time  $\geq 20$  years and an average burnup  $\leq 40,000$  MWD/MTU.

ii. All Other Array/Classes As specified in Section 2.4.

iii. NON-FUEL HARDWARE As specified in Table 2.1-8.

Table 2.1-1 (page 37 of 39)  
Fuel Assembly Limits

---

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

A. Allowable Contents (cont'd)

d. Decay Heat Per Fuel  
Storage Location:

i. Array/Classes 14x14D,  
14x14E, and 15x15G       $\leq 500$  Watts.

ii. All Other Array/Classes      As specified in Section 2.4.

e. Fuel Assembly Length       $\leq 176.8$  inches (nominal design)

f. Fuel Assembly Width       $\leq 8.54$  inches (nominal design)

g. Fuel Assembly Weight       $\leq 1,680$  lbs (including NON-FUEL  
HARDWARE)

Table 2.1-1 (page 38 of 39)  
Fuel Assembly Limits

VIII. MPC MODEL: 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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU. |
| ii. All Other Array/Classes                                       | As specified in Section 2.4.   |
| iii. NON-FUEL HARDWARE  | As specified in Table 2.1-8.   |

Table 2.1-1 (page 39 of 39)  
Fuel Assembly Limits

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

A. Allowable Contents (cont'd)

d. Decay Heat Per Fuel  
Storage Location:

i. Array/Classes 14x14D, 14x14E, and 15x15G  $\leq$  500 Watts.

ii. All Other Array/Classes As specified in Section 2.3.

e. Fuel Assembly Length  $\leq$  176.8 inches (nominal design)

f. Fuel Assembly Width  $\leq$  8.54 inches (nominal design)

g. Fuel Assembly Weight  $\leq$  1,680 lbs (including NON-FUEL  
HARDWARE and DFC)

B. Quantity per MPC: Up to eight (8) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32F fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron sources are not permitted for loading in the MPC-32F.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only be loaded in fuel storage locations 13, 14, 19 and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-2 (page 1 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	14x14E
Clad Material	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 365	≤ 412	≤ 438	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 5.0 (24E/24EF)	≤ 5.0 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Rod Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Rod Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table 2.1-2 (page 2 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 473	≤ 473	≤ 473	≤ 495	≤ 495	≤ 495
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Rod Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Rod Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 2.1-2 (page 3 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	15x15G	15x15H	16x16A	17x17A	17x17B	17x17C
Clad Material	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 495	≤ 448	≤ 433	≤ 474	≤ 480
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Note 5) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Rod Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Rod Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020



Table 2.1-2 (page 4 of 4)  
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. Deleted.
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances.
4. Each guide tube replaces four fuel rods.
5. Soluble boron concentration per LCO 3.3.1.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum initial enrichment of the INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS is 4.0 wt.% <sup>235</sup>U.

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

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 198	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 2.7	≤ 2.7 for the UO <sub>2</sub> rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Rod Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Rod Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1-3 (2 of 5)  
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 190	≤ 190	< 190	≤ 191	≤ 180
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Rod Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Rod Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

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

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 180	≤ 182	≤ 182	≤ 183	≤ 183	≤ 164
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Rod Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Rod Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

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

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 125	≤ 125
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 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
Fuel Rod Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Rod Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

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.  $\leq 0.635$  wt. %  $^{235}\text{U}$  and  $\leq 1.578$  wt. % total fissile plutonium ( $^{239}\text{Pu}$  and  $^{241}\text{Pu}$ ), (wt. % of total fuel weight, i.e.,  $\text{UO}_2$  plus  $\text{PuO}_2$ ).
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 those MPCs 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.%  $^{235}\text{U}$ , as applicable.

Table 2.1-4

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Table 2.1-5

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Table 2.1-6 (page 1 of 2)

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Table 2.1-6 (page 2 of 2)

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Table 2.1-7 (page 1 of 2)

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Table 2.1-7 (page 2 of 2)

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

<b>Post-irradiation Cooling Time (years)</b>	<b>INSERTS (Note 4) BURNUP (MWD/MTU)</b>	<b>GUIDE TUBE HARDWARE (Note 5) BURNUP (MWD/MTU)</b>	<b>CONTROL COMPONENT (Note 6) BURNUP (MWD/MTU)</b>	<b>APSR BURNUP (MWD/MTU)</b>
≥ 3	≤ 24,635	NA (Note 7)	NA	NA
≥ 4	≤ 30,000	≤ 20,000	NA	NA
≥ 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	-	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 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 TPD 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..
  5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
  6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
  7. NA means not authorized for loading at this cooling time.

## 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. A detailed discussion of how 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 Fuel Assemblies	
MPC-24	$\leq 1.157$
MPC-24E/24EF	$\leq 1.173$
MPC-32/32F	$\leq 0.898$
MPC-68/68FF	$\leq 0.414$
Damaged Fuel Assemblies and Fuel Debris	
MPC-24	$\leq 1.099$
MPC-24E/24EF	$\leq 1.114$
MPC-32/32F	$\leq 0.718$
MPC-68/68FF	$\leq 0.393$

### 2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.4-2 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in regionalized loading for each MPC model.

2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel (cont'd)

Table 2.4-2

Fuel Storage Regions and Maximum Decay Heat per MPC

MPC Model	Number of Fuel Storage Locations in Inner and Outer Regions	Inner Region Maximum Decay Heat per Assembly (kW)	Outer Region Maximum Decay Heat per Assembly (kW)
MPC-24	4 and 20	1.470	0.900
MPC-24E/24EF	4 and 20	1.540	0.900
MPC-32/32F	12 and 20	1.131	0.600
MPC-68/68FF	32 and 36	0.500	0.275

2.4.3 Burnup 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.

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

Equation 2.4.3

Where:

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

2.4.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel (cont'd)

$q$  = 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. %  $^{235}\text{U}$ )  
(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	20277.1	303.592	-68.329	-139.41	2993.67	-498.159	-615.411
≥ 4	35560.1	-6034.67	985.415	-132.734	3578.92	-723.721	-609.84
≥ 5	48917.9	-14499.5	2976.09	-150.707	4072.55	-892.691	-54.8362
≥ 6	59110.3	-22507	5255.61	-177.017	4517.03	-1024.01	613.36
≥ 7	67595.6	-30158.1	7746.6	-200.128	4898.71	-1123.21	716.004
≥ 8	74424.9	-36871.1	10169.4	-218.676	5203.64	-1190.24	741.163
≥ 9	81405.8	-44093.1	12910.8	-227.916	5405.34	-1223.27	250.224
≥ 10	86184.3	-49211.7	15063.4	-237.641	5607.96	-1266.21	134.435
≥ 11	92024.9	-55666.8	17779.6	-240.973	5732.25	-1282.12	-401.456
≥ 12	94775.8	-58559.7	19249.9	-246.369	5896.27	-1345.42	-295.435
≥ 13	100163	-64813.8	22045.1	-242.572	5861.86	-1261.66	-842.159
≥ 14	103971	-69171	24207	-242.651	5933.96	-1277.48	-1108.99
≥ 15	108919	-75171.1	27152.4	-243.154	6000.2	-1301.19	-1620.63
≥ 16	110622	-76715.2	28210.2	-240.235	6028.33	-1307.74	-1425.5
≥ 17	115582	-82929.7	31411.9	-235.234	5982.3	-1244.11	-1948.05
≥ 18	119195	-87323.5	33881.4	-233.28	6002.43	-1245.95	-2199.41
≥ 19	121882	-90270.6	35713.7	-231.873	6044.42	-1284.55	-2264.05
≥ 20	124649	-93573.5	37853.1	-230.22	6075.82	-1306.57	-2319.63

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	18937.9	70.2997	-28.6224	-130.732	2572.36	-383.393	-858.17
≥ 4	32058.7	-4960.63	745.224	-125.978	3048.98	-551.656	-549.108
≥ 5	42626.3	-10804.1	1965.09	-139.722	3433.49	-676.643	321.88
≥ 6	51209.6	-16782.3	3490.45	-158.929	3751.01	-761.524	847.282
≥ 7	57829.9	-21982	5009.12	-180.026	4066.65	-846.272	1200.45
≥ 8	62758	-26055.3	6330.88	-196.804	4340.18	-928.336	1413.17
≥ 9	68161.4	-30827.6	7943.87	-204.454	4500.52	-966.347	1084.69
≥ 10	71996.8	-34224.3	9197.25	-210.433	4638.94	-1001.83	1016.38
≥ 11	75567.3	-37486.1	10466.9	-214.95	4759.55	-1040.85	848.169
≥ 12	79296.7	-40900.3	11799.6	-212.898	4794.13	-1040.51	576.242
≥ 13	82257.3	-43594	12935	-212.8	4845.81	-1056.01	410.807
≥ 14	83941.2	-44915.2	13641	-215.389	4953.19	-1121.71	552.724
≥ 15	87228.5	-48130	15056.9	-212.545	4951.12	-1112.5	260.194
≥ 16	90321.7	-50918.3	16285.5	-206.094	4923.36	-1106.35	-38.7487
≥ 17	92836.2	-53314.5	17481.7	-203.139	4924.61	-1109.32	-159.673
≥ 18	93872.8	-53721.4	17865.1	-202.573	4956.21	-1136.9	30.0594
≥ 19	96361.6	-56019.1	19075.9	-199.068	4954.59	-1156.07	-125.917
≥ 20	98647.5	-57795.1	19961.8	-191.502	4869.59	-1108.74	-217.603

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	19176.9	192.012	-66.7595	-138.112	2666.73	-407.664	-1372.41
≥ 4	32040.3	-4731.4	651.014	-124.944	3012.63	-530.456	-890.059
≥ 5	43276.7	-11292.8	2009.76	-142.172	3313.91	-594.917	-200.195
≥ 6	51315.5	-16920.5	3414.76	-164.287	3610.77	-652.118	463.041
≥ 7	57594.7	-21897.6	4848.49	-189.606	3940.67	-729.367	781.46
≥ 8	63252.3	-26562.8	6273.01	-199.974	4088.41	-732.054	693.879
≥ 9	67657.5	-30350.9	7533.4	-211.77	4283.39	-772.916	588.456
≥ 10	71834.4	-34113.7	8857.32	-216.408	4383.45	-774.982	380.243
≥ 11	75464.1	-37382.1	10063	-218.813	4460.69	-776.665	160.668
≥ 12	77811.1	-39425.1	10934.3	-225.193	4604.68	-833.459	182.463
≥ 13	81438.3	-42785.4	12239.9	-220.943	4597.28	-803.32	-191.636
≥ 14	84222.1	-45291.6	13287.9	-218.366	4608.13	-791.655	-354.59
≥ 15	86700.1	-47582.6	14331.2	-218.206	4655.34	-807.366	-487.316
≥ 16	88104.7	-48601.1	14927.9	-219.498	4729.97	-849.446	-373.196
≥ 17	91103.3	-51332.5	16129	-212.138	4679.91	-822.896	-654.296
≥ 18	93850.4	-53915.8	17336.9	-207.666	4652.65	-799.697	-866.307
≥ 19	96192.9	-55955.8	18359.3	-203.462	4642.65	-800.315	-1007.75
≥ 20	97790.4	-57058.1	19027.7	-200.963	4635.88	-799.721	-951.122

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	15789.2	119.829	-21.8071	-127.422	2152.53	-267.717	-580.768
≥ 4	26803.8	-3312.93	415.027	-116.279	2550.15	-386.33	-367.168
≥ 5	36403.6	-7831.93	1219.66	-126.065	2858.32	-471.785	326.863
≥ 6	44046.1	-12375.9	2213.52	-145.727	3153.45	-539.715	851.971
≥ 7	49753.5	-16172.6	3163.61	-166.946	3428.38	-603.598	1186.31
≥ 8	55095.4	-20182.5	4287.03	-183.047	3650.42	-652.92	1052.4
≥ 9	58974.4	-23071.6	5156.53	-191.718	3805.41	-687.18	1025
≥ 10	62591.8	-25800.8	5995.95	-195.105	3884.14	-690.659	868.556
≥ 11	65133.1	-27747.4	6689	-203.095	4036.91	-744.034	894.607
≥ 12	68448.4	-30456	7624.9	-202.201	4083.52	-753.391	577.914
≥ 13	71084.4	-32536.4	8381.78	-201.624	4117.93	-757.16	379.105
≥ 14	73459.5	-34352.3	9068.86	-197.988	4113.16	-747.015	266.536
≥ 15	75950.7	-36469.4	9920.52	-199.791	4184.91	-779.222	57.9429
≥ 16	76929.1	-36845.6	10171.3	-197.88	4206.24	-794.541	256.099
≥ 17	79730	-39134.8	11069.4	-190.865	4160.42	-773.448	-42.6853
≥ 18	81649.2	-40583	11736.1	-187.604	4163.36	-785.838	-113.614
≥ 19	83459	-41771.8	12265.9	-181.461	4107.51	-758.496	-193.442
≥ 20	86165.4	-44208.8	13361.2	-178.89	4107.62	-768.671	-479.778

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						
	A	B	C	D	E	F	G
≥ 3	15192.5	50.5722	-12.3042	-126.906	2009.71	-235.879	-561.574
≥ 4	25782.5	-3096.5	369.096	-113.289	2357.75	-334.695	-254.964
≥ 5	35026.5	-7299.87	1091.93	-124.619	2664	-414.527	470.916
≥ 6	42234.9	-11438.4	1967.63	-145.948	2945.81	-474.981	1016.84
≥ 7	47818.4	-15047	2839.22	-167.273	3208.95	-531.296	1321.12
≥ 8	52730.7	-18387.2	3702.43	-175.057	3335.58	-543.232	1223.61
≥ 9	56254.6	-20999.9	4485.93	-190.489	3547.98	-600.64	1261.55
≥ 10	59874.6	-23706.5	5303.88	-193.807	3633.01	-611.892	1028.63
≥ 11	62811	-25848.4	5979.64	-194.997	3694.14	-618.968	862.738
≥ 12	65557.6	-27952.4	6686.74	-198.224	3767.28	-635.126	645.139
≥ 13	67379.4	-29239.2	7197.49	-200.164	3858.53	-677.958	652.601
≥ 14	69599.2	-30823.8	7768.51	-196.788	3868.2	-679.88	504.443
≥ 15	71806.7	-32425	8360.38	-191.935	3851.65	-669.917	321.146
≥ 16	73662.6	-33703.5	8870.78	-187.366	3831.59	-658.419	232.335
≥ 17	76219.8	-35898.1	9754.72	-189.111	3892.07	-694.244	-46.924
≥ 18	76594.4	-35518.2	9719.78	-185.11	3897.04	-712.82	236.047
≥ 19	78592.7	-36920.8	10316.5	-179.54	3865.84	-709.551	82.478
≥ 20	80770.5	-38599.9	11051.3	-175.106	3858.67	-723.211	-116.014

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	17038.2	158.445	-37.6008	-136.707	2368.1	-321.58	-700.033
≥ 4	29166.3	-3919.95	508.439	-125.131	2782.53	-455.722	-344.199
≥ 5	40285	-9762.36	1629.72	-139.652	3111.83	-539.804	139.67
≥ 6	48335.7	-15002.6	2864.09	-164.702	3444.97	-614.756	851.706
≥ 7	55274.9	-20190	4258.03	-185.909	3728.11	-670.841	920.035
≥ 8	60646.6	-24402.4	5483.54	-199.014	3903.29	-682.26	944.913
≥ 9	64663.2	-27753.1	6588.21	-215.318	4145.34	-746.822	967.914
≥ 10	69306.9	-31739.1	7892.13	-218.898	4237.04	-746.815	589.277
≥ 11	72725.8	-34676.6	8942.26	-220.836	4312.93	-750.85	407.133
≥ 12	76573.8	-38238.7	10248.1	-224.934	4395.85	-757.914	23.7549
≥ 13	78569	-39794.3	10914.9	-224.584	4457	-776.876	69.428
≥ 14	81559.4	-42453.6	11969.6	-222.704	4485.28	-778.427	-203.031
≥ 15	84108.6	-44680.4	12897.8	-218.387	4460	-746.756	-329.078
≥ 16	86512.2	-46766.8	13822.8	-216.278	4487.79	-759.882	-479.729
≥ 17	87526.7	-47326.2	14221	-218.894	4567.68	-805.659	-273.692
≥ 18	90340.3	-49888.6	15349.8	-212.139	4506.29	-762.236	-513.316
≥ 19	93218.2	-52436.7	16482.4	-207.653	4504.12	-776.489	-837.1
≥ 20	95533.9	-54474.1	17484.2	-203.094	4476.21	-760.482	-955.662

Table 2.4-3 (Page 7 of 8)

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

Cooling Time (years)	Array/Class 17x17A						
	A	B	C	D	E	F	G
≥ 3	16784.4	3.90244	-10.476	-128.835	2256.98	-287.108	-263.081
≥ 4	28859	-3824.72	491.016	-120.108	2737.65	-432.361	-113.457
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

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	15526.8	18.0364	-9.36581	-128.415	2050.81	-243.915	-426.07
≥ 4	26595.4	-3345.47	409.264	-115.394	2429.48	-350.883	-243.477
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953



Table 2.4-4 (Page 1 of 10)

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

Cooling Time (years)	Array/Class 7x7B						
	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						
	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

### 3.0 DESIGN FEATURES

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#### 3.1 Site

##### 3.1.1 Site Location

The HI-STORM 100 Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

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#### 3.2 Design Features Important for Criticality Control

##### 3.2.1 MPC-24

1. Flux trap size:  $\geq 1.09$  in.
2.  $^{10}\text{B}$  loading in the neutron absorbers:  $\geq 0.0267$  g/cm<sup>2</sup> (Boral) and  $\geq 0.0223$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.2 MPC-68 and MPC-68FF

1. Fuel cell pitch:  $\geq 6.43$  in.
2.  $^{10}\text{B}$  loading in the neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) and  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.3 MPC-68F

1. Fuel cell pitch:  $\geq 6.43$  in.
2.  $^{10}\text{B}$  loading in the Boral neutron absorbers:  $\geq 0.01$  g/cm<sup>2</sup>

##### 3.2.4 MPC-24E and MPC-24EF

1. Flux trap size:
  - i. Cells 3, 6, 19, and 22:  $\geq 0.776$  inch
  - ii. All Other Cells:  $\geq 1.076$  inches
2.  $^{10}\text{B}$  loading in the neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) and  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |

##### 3.2.5 MPC-32 and MPC-32F

1. Fuel cell pitch:  $\geq 9.158$  inches
2.  $^{10}\text{B}$  loading in the neutron absorbers:  $\geq 0.0372$  g/cm<sup>2</sup> (Boral) and  $\geq 0.0310$  g/cm<sup>2</sup> (METAMIC) |

## DESIGN FEATURES

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### 3.2 Design features Important for Criticality Control (cont'd)

3.2.6 The fuel assemblies shall be positioned in the MPC to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.

3.2.7 The  $B_4C$  content in METAMIC shall be  $\leq 33.0$  wt.%.

#### 3.2.8 Neutron Absorber Tests

Section 9.1.5.3 of the HI-STORM 100 FSAR is hereby incorporated by reference into the HI-STORM 100 CoC. The minimum  $^{10}B$  for the neutron absorber shall meet the minimum requirements for each MPC model specified in Sections 3.2.1 through 3.2.5 above.

### 3.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing Code for the HI-STORM 100 System, as clarified in Specification 3.3.1 below, except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections, provided a written reconciliation of the later edition against the 1995 Edition, including addenda, is performed by the certificate holder. American Concrete Institute (ACI) 349-85 is the governing Code for plain concrete as clarified in Appendix 1.D of the Final Safety Analysis Report for the HI-STORM 100 Cask System.

#### 3.3.1 Alternatives to Codes, Standards, and Criteria

Table 3-1 lists approved alternatives to the ASME Code for the design of the HI-STORM 100 Cask System.

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

Proposed alternatives to the ASME Code, Section 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

(continued)

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## DESIGN FEATURES

3.3.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria (cont'd)

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. |

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(continued)

DESIGN FEATURES

Table 3-1 (page 1 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

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

Table 3-1 (page 2 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC basket supports and lift lugs	NB-1130	<p>NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material.</p> <p>NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.</p>	<p>The MPC basket supports (nonpressure-retaining structural attachments) and lift lugs (nonstructural attachments (relative to the function of lifting a loaded MPC) that are used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.</p>
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.

Table 3-1 (page 3 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC, MPC basket assembly, HI-STORM OVERPACK and HI-TRAC TRANSFER CASK	NB-3100 NG-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.
MPC	NB-3350	NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.	<p>Due to MPC basket-to-shell interface requirements, the MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.</p> <p>From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.</p>



Table 3-1 (page 4 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC, MPC Basket Assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	NB-4120 NG-4120 NF-4120	NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.	<p>In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of lead are not, unless explicitly stated by the Code, defined as heat treatment operations.</p> <p>For the steel parts in the HI-STORM 100 System components, the duration for which a part exceeds the off-normal temperature limit defined in Chapter 2 of the FSAR shall be limited to 24 hours in a particular manufacturing process (such as the HI-TRAC lead pouring process).</p>
MPC, MPC basket assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	NB-4220 NF-4220	Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.	<p>The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-TRANSFER CASK) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.</p>
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3).	<p>MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.</p>

Table 3-1 (page 5 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	<p>The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be pressure tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant pressure test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded.</p> <p>The inspection results, including relevant findings (indications), shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.</p>

Table 3-1 (page 6 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. The function of the MPC enclosure vessel is to contain the radioactive contents under normal, off-normal, and accident conditions. The MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NG-2000 requirements.

Table 3-1 (page 7 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC basket assembly	NG-4420	NG-4427(a) allows a fillet weld in any single continuous weld to be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. The MPC basket data package to be in accordance with Holtec approved QA program.
OVERPACK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.

Table 3-1 (page 8 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
TRANSFER CASK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.
OVERPACK Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member of thickness 1 inch or greater is loaded to transmit loads in the through thickness direction.	The margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The OVERPACK baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have large safety factors during lifting. Likewise, the top lid plate to lid shell weld has a large structural margin under the inertia loads imposed during a non-mechanistic tipover event.
OVERPACK Steel Structure	NF-3256 NF-3266	Provides requirements for welded joints.	<p>Welds for which no structural credit is taken are identified as "Non-NF" welds in the design drawings. These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX.</p> <p>Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.</p>

Table 3-1 (page 9 of 9)  
LIST OF ASME CODE ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
HI-STORM OVERPACK and HI-TRAC TRANSFER CASK	NF-3320 NF-4720	NF-3324.6 and NF-4720 provide requirements for bolting	<p>These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The OVERPACK and TRANSFER CASK bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).</p> <p>Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.</p>

DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

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

1. The temperature of 80° F is the maximum average yearly temperature.
2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.
3. a. For free-standing casks, the resultant horizontal acceleration (vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site),  $G_H$ , and vertical ZPA,  $G_V$ , expressed as fractions of 'g', shall satisfy the following inequality:

$$G_H + \mu G_V \leq \mu$$

where  $\mu$  is either the Coulomb friction coefficient for the HI-STORM 100/ISFSI pad interface or the ratio  $r/h$ , where 'r' is the radius of the cask and 'h' is the height of the cask center-of-gravity above the ISFSI pad surface. The above inequality must be met for both definitions of  $\mu$ . Unless demonstrated by appropriate testing that a higher coefficient of friction value is appropriate for a specific ISFSI, the value used shall be 0.53. Representative values of  $G_H$  and  $G_V$  combinations for a coefficient of friction = 0.53 to prevent sliding are provided in Table 3-2. If acceleration time-histories on the ISFSI pad surface are available,  $G_H$  and  $G_V$  may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequality shall be evaluated at each time step in the acceleration time history over the total duration of the seismic event.

Table 3-2

Representative DBE Acceleration Values to Prevent HI-STORM 100 Sliding ( $\mu = 0.53$ )

<b>Equivalent Vectorial Sum of Two Horizontal ZPA's (<math>G_H</math> in g's)</b>	<b>Corresponding Vertical ZPA (<math>G_V</math> in g's)</b>
0.445	0.160
0.424	0.200
0.397	0.250

(continued)

DESIGN FEATURES

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3.4 Site-Specific Parameters and Analyses (continued)

- b. For free-standing casks, under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will not result in cask tipover or cause a cask to fall off the pad. In addition, impact between casks should be precluded, or should be considered an accident for which the maximum g-load experienced by the stored fuel shall be limited to 45 g's.
- c. For those ISFSI sites with design basis seismic acceleration values higher than those allowed for free-standing casks, the HI-STORM 100 System shall be anchored to the ISFSI pad. The site seismic characteristics and the anchorage system shall meet the following requirements:
  - i. The site acceleration response spectra at the top of the ISFSI pad shall have ZPAs that meet the following inequalities:

$$G_H \leq 2.12$$

AND

$$G_V \leq 1.5$$

Where:

$G_H$  is the vectorial sum of the two horizontal ZPAs at a three-dimensional seismic site (or the horizontal ZPA at a two-dimensional site) and  $G_V$  is the vertical ZPA.

- ii. Each HI-STORM 100 dry storage cask shall be anchored with twenty-eight (28), 2-inch diameter studs and compatible nuts of material suitable for the expected ISFSI environment. The studs shall meet the following requirements:

Yield Strength at Ambient Temperature:  $\geq 80$  ksi

Ultimate Strength at Ambient Temperature:  $\geq 125$  ksi

Initial Tensile Pre-Stress:  $\geq 55$  ksi AND  $\leq 65$  ksi

NOTE: The above anchorage specifications are required for the seismic

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## DESIGN FEATURES

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### 3.4 Site-Specific Parameters and Analyses (continued)

spectra defined in item 3.4.3.b.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:  $\geq 4,000$  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 shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.
- 6.
  - a. For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to  $\leq 45$  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 lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having 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  $\leq 45$  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 lift height above the ISFSI pad is not required to be established if the cask is lifted with a device design in accordance with ANSI N14.6 and having redundant drop protection features.
- 7. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

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(continued)

DESIGN FEATURES

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3.4 Site-Specific Parameters and Analyses (continued)

8. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures  $\geq 0^{\circ}$  F.
9. For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e, longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.

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(continued)

## DESIGN FEATURES

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### 3.5 Cask Transfer Facility (CTF)

#### 3.5.1 TRANSFER CASK and MPC Lifters

Lifting of a loaded TRANSFER CASK and MPC using devices that are not integral to structures governed by 10 CFR Part 50 shall be performed with a CTF that is designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and the below clarifications. The CTF Structure requirements below do not apply to heavy loads bounded by the regulations of 10 CFR Part 50.

#### 3.5.2 CTF Structure Requirements

##### 3.5.2.1 Cask Transfer Station and Stationary Lifting Devices

1. The metal weldment structure of the CTF structure shall be designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. The applicable loads, load combinations, and associated service condition definitions are provided in Table 3-3. All compression loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF.
2. If a portion of the CTF structure is constructed of reinforced concrete, then the factored load combinations set forth in ACI-318 (89) for the loads defined in Table 3-3 shall apply.
3. The TRANSFER CASK and MPC lifting device used with the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1.
4. The CTF shall be designed, constructed, and evaluated to ensure that if the MPC is dropped during inter-cask transfer operations, its confinement boundary would not be breached. This requirements applies to CTFs with either stationary or mobile lifting devices.

(continued)

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DESIGN FEATURES

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3.5.2.2 Mobile Lift Devices

If a mobile lifting device is used as the lifting device, in lieu of a stationary lifting device, it shall meet the guidelines of NUREG-0612, Section 5.1, with the following clarifications:

1. Mobile lifting devices shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6(1)(a) and shall be capable of stopping and holding the load during a Design Basis Earthquake (DBE) event.
2. Mobile lifting devices shall conform to meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes."
3. Mobile cranes are not required to meet the requirements of NUREG-0612, Section 5.1.6(2) for new cranes.
4. Horizontal movements of the TRANSFER CASK and MPC using a mobile crane are prohibited.

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(continued)

DESIGN FEATURES

Table 3-3

Load Combinations and Service Condition Definitions for the CTF Structure (Note 1)

Load Combination	ASME III Service Condition for Definition of Allowable Stress	Comment
D* D + S	Level A	All primary load bearing members must satisfy Level A stress limits
D + M + W' (Note 2) D + F D + E D + Y	Level D	Factor of safety against overturning shall be $\geq 1.1$

D = Dead load  
D\* = Apparent dead load  
S = Snow and ice load for the CTF site  
M = Tornado missile load for the CTF site  
W' = Tornado wind load for the CTF site  
F = Flood load for the CTF site  
E = Seismic load for the CTF site  
Y = Tsunami load for the CTF site

- Notes:
1. The reinforced concrete portion of the CTF structure shall also meet the factored combinations of loads set forth in ACI-318(89).
  2. Tornado missile load may be reduced or eliminated based on a PRA for the CTF site.

## DESIGN FEATURES

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### 3.6 Forced Helium Dehydration System

#### 3.6.1 System Description

Use of the Forced Helium Dehydration (FHD) system, (a closed-loop system) is an alternative to vacuum drying the MPC for moderate burnup fuel ( $\leq 45,000$  MWD/MTU) and mandatory for drying MPCs containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.6.2.

#### 3.6.2 Design Criteria

- 3.6.2.1 The temperature of the helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.
- 3.6.2.2 The pressure in the MPC cavity space shall be  $\leq 60.3$  psig (75 psia).
- 3.6.2.3 The hourly recirculation rate of helium shall be  $\geq 10$  times the nominal helium mass backfilled into the MPC for fuel storage operations.
- 3.6.2.4 The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. The limit is met if the gas temperature at the demoisurizer outlet is verified by measurement to remain  $\leq 21^\circ\text{F}$  for a period of 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain  $\leq 22.9^\circ\text{F}$  for  $\geq 30$  minutes.
- 3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point  $\leq 120^\circ\text{F}$ .
- 3.6.2.6 The demoisurizing module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in FSAR Appendix 2.B) has been completed.
- 3.6.2.7 The helium circulator shall be sized to effect the minimum flow rate of circulation required by these design criteria.
- 3.6.2.8 The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets these design criteria.

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(continued)

DESIGN FEATURES

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3.6 Forced Helium Dehydration System (continued)

3.6.3 Fuel Cladding Temperature

A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Section 4.4, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation, is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

**LAR 1014-2, REVISION 2**

**ATTACHMENT 4**

**PROPOSED FSAR CHANGES**



**FINAL SAFETY ANALYSIS REPORT**

**for the**

**HOLTEC INTERNATIONAL**

**STORAGE, AND TRANSFER**

**OPERATION REINFORCED MODULE**

**CASK SYSTEM**

**(HI-STORM 100 CASK SYSTEM)**

**DOCKET 72-1014**

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## CHAPTER 1<sup>†</sup>: GENERAL DESCRIPTION

### 1.0 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) for Holtec International's HI-STORM 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under requirements specified in 10CFR72 [1.0.1]. This FSAR describes the basis for NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L, for the HI-STORM 100 System to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI). This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process.

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

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

The HI-STORM 100 FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all guidance. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.3 herein. Table 1.0.3 also contains the justification for the alternative method for compliance adopted in this FSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

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

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

- Siting of the ISFSI and design of the storage pad (including the embedment for anchored cask users) and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.
- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.
- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.
- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.
- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.
- Performance of pre-operational testing.
- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.
- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

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

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

Revisions to this document are made on a section level basis. Complete sections have been replaced if any material in the section changed. The specific changes are noted with revision bars in the right margin. Figures are revised individually. Drawings are controlled separately within the Holtec QA program and have individual revision numbers. Bills-of-Material (BOMs) are considered separate drawings and are not necessarily at the same revision level as the drawing(s) to which they apply. If a drawing or BOM was revised in support of the current FSAR revision, that drawing/BOM is included in Section 1.5 at its latest revision level. Drawings and BOMs appearing in this FSAR may be revised between formal updates to the FSAR. Therefore, the revisions of drawings/BOMs in Section 1.5 may not be current.

#### 1.0.1 Engineering Change Orders

The changes authorized by the following Holtec Engineering Change Orders (ECOs) are reflected in Revision 1 of this FSAR:

MPC-68/68F/68FF: ECOs 1021-1 through 4, 7, 8, 12 through 16, 18 through 23, 27 through 30, 33, 34, 36, 38, 39, 41, 43, and 44; and 71188-43.

MPC-24/24E/24EF: ECOs 1022- 1 through 7, 9, 10, 12 through 26, 28, 31, and 34 through 38.

MPC-32: ECOs 1023-1 and 3 through 10.

HI-STORM overpack: ECOs 1024-1 through 4, 6 through 16, 18 through 21, 24, 25, 27 through 38, 42 through 47, 50, 51, 52, 54 through 58, and 60.

HI-TRAC 125 transfer cask: ECOs 1025-1 through 32, 35, and 36.

HI-TRAC 100 transfer cask: ECOs 1026- 1 through 29.

Ancillary Equipment: ECOs 1027-27, 31, 33, 46, and 53.

General FSAR changes: ECOs 5014-36, 47, 49, 50, 51, 53, 54, 56, 58 through 64, 66, 67, and 68.

**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

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

**Boral** is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

**Boral<sup>TM</sup>** means Boral manufactured by AAR Advanced Structures.

**BWR** is an acronym for boiling water reactor.

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

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

**Confinement Boundary** means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

**Confinement System** means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

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

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

**DBE** means Design Basis Earthquake.

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

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

**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

**Damaged Fuel Container (or Canister)** means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The Damaged Fuel Container/Canister (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC.

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

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

**Design Report** is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. *The FSAR serves as the Design Report for the HI-STORM 100 System.*

**Design Specification** is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM 100 System. *The FSAR serves as the Design Specification for the HI-STORM 100 System.*

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

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

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

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

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

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

**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

**HI-TRAC transfer cask or HI-TRAC** means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. HI-TRAC is an acronym for **Holtec International Transfer Cask**. In this FSAR there are three HI-TRAC transfer casks, the 125 ton standard design HI-TRAC (HI-TRAC-125), the 125-ton dual-purpose lid design (HI-TRAC 125D), and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to all three HI-TRAC transfer cask design, unless the discussion requires distinguishing among the three. The HI-TRAC is equipped with a pair of lifting trunnions and the HI-TRAC 100 and HI-TRAC 125 designs also include pocket trunnions. The trunnions are used to lift and downend/upend the HI-TRAC with a loaded MPC.

**HI-STORM overpack** or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. The term “overpack” as used in this FSAR refers to both the standard design overpack (HI-STORM 100), the alternate design overpack (HI-STORM 100S), and either of these as an overpack designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA), unless otherwise clarified.

**HI-STORM 100 System** consists of a loaded MPC placed within the HI-STORM ~~100~~ overpack. |

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

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

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

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

**Intact Fuel Assembly** is defined as a fuel assembly without known or suspected cladding defects

**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

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

***Long-term Storage** means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad and lasting up to the end of the licensed life of the HI-STORM 100 System (20 years).*

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

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

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

**METCON<sup>™</sup>** is a trade name for the HI-STORM 100 overpack. The trademark is derived from the metal-concrete composition of the HI-STORM 100 overpack.

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

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

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

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

**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

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

***Neutron Absorber Material** is a generic term used in this FSAR to indicate any neutron absorber material qualified for use in the HI-STORM 100 System MPCs.*

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

**Non-Fuel Hardware** is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), water displacement guide tube plugs, and-orifice rod assemblies, and vibration suppressor inserts.

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

**Plain Concrete** is concrete that is unreinforced and is of density specified in this FSAR .

~~**Preferential Fuel Loading** is a requirement in the CoC applicable to uniform fuel loading whenever fuel assemblies with significantly different post irradiation cooling times ( $\geq 1$  year) are to be loaded in the same MPC. Fuel assemblies with the longest post irradiation cooling time are loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post irradiation cooling times are placed toward the center of the basket. Regionalized fuel loading meets the intent of preferential fuel loading. Preferential fuel loading is a requirement in addition to other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers.~~

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

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

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

**Regionalized Fuel Loading** is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high heat emitting fuel assemblies to be stored in fuel storage locations in the center of the fuel basket provided lower heat emitting fuel assemblies are stored in the peripheral fuel storage locations. Users choosing regionalized fuel loading must also consider other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers. ~~Regionalized fuel loading meets the intent of preferential fuel loading.~~ *Regionalized fuel loading does not apply to the MPC-68F model.*

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



**Table 1.0.1**

**TERMINOLOGY AND NOTATION**

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

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

**Single Failure Proof** means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

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

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

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

*Thermal Capacity of the HI-STORM system is defined as the amount of heat the storage system, containing an MPC loaded with CSF stored in uniform storage, will actually reject with the ambient environment at the normal temperature and the peak fuel cladding temperature (PCT) at 400°C.*

**Thermosiphon** is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket. *maximum heat load during short-term operating conditions up to which no time limit or other restriction is imposed on the operating condition.*

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

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

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

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

Table 1.0.2

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX

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

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.5
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)(2)	2.2.3.6
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b)(2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b)(2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a)(4)	2.3.2.2, 4.0
NA	2.III.3f Operating Procedures	10CFR72.24(f) 10CFR72.128(a)(5)	10.0, 8.0
		10CFR72.236(h)	8.0
		10CFR72.24(1)(2)	1.2.1, 1.2.2
		10CFR72.236(1)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a)(1)	9.0

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a) (2)	2.3.5.2
		10CFR72.128(a) (3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
10CFR72.236(e)	2.3.2.1		
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h) (4) 10CFR72.122(i) 10CFR72.128(a) (1)	2.3.5
2.3.4 Nuclear Criticality Safety	2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236(h)	2.4
	14.III.1 Design	10CFR72.130	2.4
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(1)
	14.III.4 License Termination	10CFR72.54	(1)
<b>3. Structural Evaluation</b>			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.24(c)	3.1
3.2 Weights and Centers of Gravity	3.V.1.b.2 Structural Design Features	--	3.2
3.3 Mechanical Properties of Materials	3.V.1.c Structural Materials	10CFR72.24(c)(3)	3.3
	3.V.2.c Structural Materials		
NA	3.III.2 Radiation Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(1)	3.4.4.3 3.4.7.3 3.4.10
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4.3
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.102(f)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.236(g)	3.4.11 3.4.12
3.4 General Standards for Casks	--	--	3.4
3.4.1 Chemical and Galvanic Reactions	3.V.1.b.2 Structural Design Features	--	3.4.1
3.4.2 Positive Closure	--	--	3.4.2
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions --	--	3.4.3, Appendices 3.E, 3.AC, 3.D

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4, Appendices 3.I, 3.U, 3.V, 3.W
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h) (1)	3.5
<b>4. Thermal Evaluation</b>			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a) (4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria <i>ISG-11, Revision 3</i>	10CFR72.122(h) (1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria <i>ISG-11, Revision 3</i>	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
<b>5. Shielding Evaluation</b>			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configurations	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a) (2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
<b>6. Criticality Evaluation</b>			
6.1 Discussion and Results	--	--	6.1
6.2 Spent Fuel Loading	6.V.2 Fuel Specification	--	6.1, 6.2
6.3 Model Specifications	6.V.3 Model Specification	--	6.3
6.3.1 Description of Calculational Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2, 6.3.3
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6 Supplemental Data	6.V.5 Supplemental Info.	--	Appendices 6.B,6.C, and 6.D
<b>7. Confinement</b>			
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety <i>ISG-18</i>	10CFR72.24(c)(3) 10CFR72.24(1)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h) (l)	7.1, 7.1.1, <del>7.2.2</del>
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System <i>ISG-18</i>	10CFR72.24(d) 10CFR72.236(1)	<del>7.2</del> <i>1</i>
7.2.1 Release of Radioactive	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	<del>7.2</del> <i>1</i>



**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
Material	7.III.4 Monitoring of Confinement System	10CFR72.122(h) (4) 10CFR72.128(a) (I)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose <i>ISG-18</i>	10CFR72.104(a)	<del>7.3.5</del> 7.1
7.2.2 Pressurization of Confinement Vessel	--	--	<del>7.2.2</del> 7.1
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System <i>ISG-18</i>	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	<del>7.3</del> 7.1
7.3.1 Fission Gas Products	--	--	<del>7.3.1</del> 7.1
7.3.2 Release of Contents	<i>-ISG-18</i>	--	<del>7.3.3</del> 7.1
NA	--	10CFR72.106(b)	<del>7.3</del> 7.1
7.4 Supplemental Data	7.V Supplemental Info.	--	--
<b>8. Operating Procedures</b>			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.1.5, 8.5.2
	8.III.4 Written Procedures	10CFR72.212(b) (9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0 Introduction

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

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

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1 <sup>(2)</sup>
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a) (1)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b) (8)	9.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	<sup>(3)</sup>
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	<sup>(4)</sup>
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	9.1.7, 9.1.1.(12)
	9.III.3 Cask Identification		
<b>10. Radiation Protection</b>			
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	10.2

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI-STORM FSAR</b>
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
<b>11. Accident Analyses</b>			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	8.3

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
<b>12. Operating Controls and Limits</b>			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart E 10CFR72 Subpart F	12.0
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 12.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 12.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

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

**Table 1.0.2 (continued)**

**HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE  
CROSS REFERENCE MATRIX**

Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
<b>13. Quality Assurance</b>			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) <i>10CFR72.140(d)</i>	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Notes:

- (1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application.
- (2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission’s acceptance of the tests defined in Chapter 9.
- (3) Not applicable to HI-STORM 100 System. The functional adequacy of all important to safety components is demonstrated by analyses.
- (4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.
- (5) The stated requirement is not applicable to the HI-STORM 100 System. No monitoring is required for accident conditions.
- “—” There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.
- “NA” There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>2.V.2.(b)(1) "The NRC accepts as the maximum and minimum "normal" temperatures the highest and lowest ambient temperatures recorded in each year, averaged over the years of record."</p>	<p><u>Exception:</u> Section 2.2.1.4 for environmental temperatures utilizes an upper bounding value of 80°F on the annual average ambient temperatures for the United States.</p>	<p>The 80°F temperature set forth in Table 2.2.2 is greater than the annual average ambient temperature at any location in the continental United States. Inasmuch as the primary effect of the environmental temperature is on the computed fuel cladding temperature to establish long-term fuel cladding integrity, the annual average ambient temperature for each ISFSI site should be below 80°F. The large thermal inertia of the HI-STORM 100 System ensures that the daily fluctuations in temperatures do not affect the temperatures of the system. Additionally, the 80°F ambient temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours.</p>
<p>2.V.2.(b)(3)(f) "10CFR Part 72 identifies several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for spent fuel storage."</p>	<p><u>Clarification:</u> A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM 100 System must be performed prior to use if these events are applicable to the site.</p>	<p>In accordance with NUREG-1536, 2.V.(b)(3)(f), if seiche, tsunami, and hurricane are not addressed in the SAR and they prove to be applicable to the site, a safety analysis is required prior to approval for use of the DCSS under either a site specific, or general license.</p>



Table 1.0.3 (continued)

## HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."</p>	<p><u>Clarification:</u> As stated in NUREG-1536, 3.V.(d), page 3-11, "Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted outside the spent fuel building, or the maximum deceleration that the cask could experience in a drop." The maximum deceleration for a corner drop is specified as 45g's for the HI-STORM overpack. No carry height limit is specified for the corner drop.</p>	<p>In Chapter 3, the MPC and HI-STORM overpack are evaluated under a 45g radial loading. A 45g axial loading on the MPC is bounded by the analysis presented in the HI-STAR FSAR, Docket 72-1008, under a 60g loading, and is not repeated in this FSAR. In Chapter 3, the HI-STORM overpack is evaluated under a 45g axial loading. Therefore, the HI-STORM overpack and MPC are qualified for a 45g loading as a result of a corner drop. Depending on the design of the lifting device, the type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses may be required to demonstrate that the deceleration limit of 45g's is not exceeded.</p>
<p>3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced..."</p> <p>3.V.2.b.i.(2)(b), Page 3-20, Para. 1, "The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359".</p> <p>3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and construction of reinforced concrete structures important to safety but not within the scope of ACI 359 should comply with the requirements of ACI 349".</p>	<p><u>Exception:</u> The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used for the material selection and specification, and construction of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack. ACI 318-95 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.</p>	<p>Concrete is provided in the HI-STORM overpack solely for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose.</p> <p>To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. <i>The temperature limits for normal conditions are per Paragraph A.4.3 of Appendix A to ACI 349 and temperature limits for</i></p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
		<p><i>off-normal and accident conditions are per Paragraph A.4.2 of Appendix A to ACI 349. In addition, the temperature limits for normal and off-normal condition from ACI 349 will be imposed.</i></p> <p>Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.</p>
<p>3.V.3.b.i.(2), Page 3-29, Para. 1, "The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with Section III of the ASME B&amp;PV Code."</p>	<p><u>Clarification:</u> The HI-STORM overpack steel structure is designed in accordance with the ASME B&amp;PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.</p>	<p>The overpack structure is a steel weldment consisting of "plate and shell type" members. As such, it is appropriate to design the structure to Section III, Class 3 of Subsection NF. The very same approach has been used in the structural evaluation of the "intermediate shells" in the HI-STAR 100 overpack (Docket Number 72-1008) previously reviewed and approved by the USNRC.</p>
<p>4.V.5, Page 4-2 "for each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage."</p> <p>4.V.1, Page 4-3, Para. 1 "the staff should verify that cladding temperatures for each fuel type proposed for storage will be below the expected damage thresholds for normal conditions of storage."</p> <p>4.V.1, Page 4-3, Para. 2 "fuel cladding limits for each fuel type should be defined in the SAR with thermal restrictions in the DCSS technical specifications."</p>	<p><u>Clarification:</u> As described in Section 4.3, all fuel array types authorized for storage <i>are assigned a single peak fuel cladding temperature limit.</i> <del>have been evaluated for the peak fuel cladding temperature limit.</del></p>	<p>As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak <i>normal</i> fuel cladding temperature limit of <i>400°C</i>. <del>All major variations in fuel parameters are considered in the determination of the peak fuel cladding temperature limits. Minor variations in fuel parameters within an array type are bounded by the conservative determination of the peak fuel cladding temperature limit.</del></p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>4.V.1, Page 4-3, Para. 4 "the applicant should verify that these cladding temperature limits are appropriate for all fuel types proposed for storage, and that the fuel cladding temperatures will remain below the limit for facility operations (e.g., fuel transfer) and the worst-case credible accident."</p>		
<p>4.V.4.a, Page 4-6, Para. 3 "applicants seeking NRC approval of specific internal convection models should propose, in the SAR, a comprehensive test program to demonstrate the adequacy of the cask design and validation of the convection models."</p>	<p><u>Exception:</u> <del>The natural convection model described in Subsection 4.4.1. is based on classical correlations for natural convection in differentially heated cavities which have been validated by many experimental studies. Therefore,</del> a No additional test program is proposed.</p>	<p><i>The HI-STORM System FLUENT computational fluid dynamics model has been benchmarked against data from an in-service spent fuel storage cask and very good agreement was found. Many experimental studies of this mechanism have been performed by others and reported in open literature sources. As discussed in Subsection 4.4.1, natural convection has been limited to the relatively large MPC basket to shell peripheral gaps. Subsection 4.4.1 provides sufficient references to experiments which document the validity of the classical correlation used in the analysis.</i></p>
<p>4.V.4.a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."</p>	<p><u>Clarification:</u> As discussed in Subsection 4.4.2, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based update of Wooten-Epstein (described in Subsection 4.4.1.1.2)</p>	<p>The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature difference conservatively maximizes the basket wall temperature.</p>

Table 1.0.3 (continued)

## HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
4.V.4.b, Page 4-7, Para. 2 "if the thermal model is axisymmetric or three dimensional, the longitudinal thermal conductivity should generally be limited to the conductivity of the cladding (weighted fractional area) within the fuel assembly."	<u>Clarification:</u> As described in Subsection 4.4.1.1.4, the axial thermal conductivity of the fuel basket is set equal to the cross-sectional thermal conductivity.	Due to the large number of gaps in the cross-sectional heat transfer paths, use of the fuel basket cross-sectional thermal conductivity for the axial thermal conductivity severely underpredicts the axial thermal conductivity of the fuel basket region. This imposed axial thermal conductivity restriction is even more limiting than that imposed by this requirement of NUREG-1536.
4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."	<u>Exception:</u> All calculations of fuel assembly effective thermal conductivities, described in Subsection 4.4.1.1.2, use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.	Within Subsection 4.4.1.1.2, the calculated effective thermal conductivities based on nominal design fuel dimensions are compared with available literature values and are demonstrated to be conservative by a substantial margin.
4.V.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented."	<u>Clarification:</u> No additional heat balance is performed or provided.	The FLUENT computational fluid dynamics program used to perform evaluations of the HI-STORM Overpack and HI-TRAC transfer cask, which uses a discretized numerical solution algorithm, enforces an energy balance on all discretized volumes throughout the computational domain. This solution method, therefore, ensures a heat balance at the surface of the cask.
4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."	<u>Exception:</u> No input or output file listings are provided in Chapter 4.	A complete set of computer program input and output files would be in excess of three hundred pages. All computer files are considered proprietary because they provide details of the design and analysis methods. In order to minimize the amount of proprietary information in the FSAR, computer files are provided in the proprietary calculation packages.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures.</p>	<p><u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) are calculated using maximum weights and minimum densities.</p>	<p>Calculating the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) using maximum weights and minimum densities conservatively overpredicts the volume occupied by the internal components and correspondingly underpredicts the remaining free volume.</p>
<p><del>7.V.4.c, Page 7-7, Para. 2 and 3 "Because the leak is assumed to be instantaneous, the plume meandering factor of Regulatory Guide 1.145 is not typically applied." and "Note that for an instantaneous release (and instantaneous exposure), the time that an individual remains at the controlled area boundary is not a factor in the dose calculation."</del>            7.V.4 "Confinement Analysis. Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary."</p>	<p><del><u>Exception:</u> As described in Section 7.3, in lieu of an instantaneous release, the assumed leakage rate is set equal to the leakage rate acceptance criteria (<math>5 \times 10^{-6}</math> atm-cm<sup>3</sup>/s) plus 50% for conservatism, which yields <math>7.5 \times 10^{-6}</math> atm-cm<sup>3</sup>/s. Because the release is assumed to be a leakage rate, the individual is assumed to be at the controlled area boundary for 720 hours. Additionally, the atmospheric dispersion factors of Regulatory Guide 1.145 are applied. No confinement analysis is performed and no effluent dose at the controlled area boundary is calculated.</del></p>	<p>The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g., <del>helium leakage, hydrostatic, and volumetric weld inspection</del> <i>non-destructive examination, pressure testing, and fabrication shop leakage testing</i>).</p> <p><i>Pursuant to ISG-18, the Holtec MPC is constructed in a manner that supports leakage from the confinement boundary being non-credible. Therefore, no confinement analysis is required.</i></p>
<p>9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."</p>	<p><u>Clarification:</u> Section 9.1.1.1 and the Design Drawings specify that the ASME Code, Section III, Subsection NB, Article NB-5332 will be used for the acceptance criteria for the volumetric examination of the MPC lid-to-shell weld.</p>	<p>In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria as defined by either the ASME code, or an alternative approach..." The ASME Code, Section III, Subsection NB, Paragraph NB-5332 is appropriate acceptance criteria for pre-service examination.</p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p>9.V.1.d, Para. 1 "Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron absorbing material."</p>	<p><u>Exception:</u> Subsection 9.1.5 describes the control of special processes, such as neutron shield material installation, to be performed in lieu of scanning or probing with neutron sources.</p>	<p>The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation.</p> <p>The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures.</p> <p>The composition of the Holtite-A is confirmed by inspection and tests prior to first use.</p> <p>Following the first loading for the HI-TRAC transfer cask and each HI-STORM overpack, a shield effectiveness test is performed in accordance with written approved procedures, as specified in Section 9.1.</p>
<p>13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."</p>	<p><u>Exception:</u> Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.</p>	<p><i>The NRC has approved Revision 13 of the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec will apply this QA program to all important-to-safety dry storage cask activities. Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation.</i></p>

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
<p><del>ISG-15, Section X.5.4.2, "No more than 1% of the rods in an assembly have peak cladding oxide thicknesses greater than 80 micrometers and no more than 3% of the rods in an assembly have peak cladding oxide thicknesses greater than 70 micrometers. A high burnup fuel assembly should be treated as potentially damaged fuel if the assembly does not meet both of the above criteria or if the fuel assembly contains fuel rods with oxide that has become detached or spalled from the cladding.</del></p>	<p><del>The Fuel Cladding Oxide Thickness Evaluation Program in Section 5.0 of Appendix A to the CoC provides an equation to calculate the maximum allowable high burnup fuel cladding oxide thickness, based on fuel assembly type.</del></p>	<p>FSAR Appendix 4.A, Section 4.A.9 provides the justification for this deviation from NUREG-1536 (ISG-15).</p>

## 1.1 INTRODUCTION

HI-STORM 100 (acronym for Holtec International Storage and Transfer Operation Reinforced Module) is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10CFR72. The annex "100" is a model number designation which denotes a system weighing over 100 tons. The HI-STORM 100 System consists of a sealed metallic canister, herein abbreviated as the "MPC", contained within an overpack. Its design features are intended to simplify and reduce on-site SNF loading, handling, and monitoring operations, and to provide for radiological protection and maintenance of structural and thermal safety margins.

The HI-STORM 100S overpack is a variant of the HI-STORM 100 overpack and has its own set of drawings in Section 1.5. The "S" suffix indicates an enhanced overpack design, as described later in this section. The HI-STORM 100S accepts the same MPCs and fuel types as the HI-STORM 100 and the basic structural, shielding, and thermal-hydraulic characteristics remain unchanged. Hereafter in this FSAR reference to HI-STORM 100 System or the HI-STORM overpack is construed to apply to both the HI-STORM 100 and the HI-STORM 100S. Where necessary, the text distinguishes between the two overpack designs. See Figures 1.1.1A and 1.1.3A for a pictorial view of the HI-STORM 100S overpack design.

The HI-STORM 100A overpack is a third variant of the HI-STORM 100 family and is specially outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the ISFSI pad in high seismic applications. In the following, the modified structure of the HI-STORM 100A, in each of four quadrants, is denoted as a "sector lug." The HI-STORM 100A design is also applicable to the HI-STORM 100S overpack, in which case the assembly would be named HI-STORM 100SA. Hereafter in the text, discussion of HI-STORM 100A applies to both the standard (HI-STORM 100A) and short (HI-STORM 100SA) overpacks, unless otherwise clarified.

The HI-STORM 100 System is designed to accommodate a wide variety of spent nuclear fuel assemblies in a single overpack design by utilizing different MPCs. The external dimensions of all MPCs are identical to allow the use of a single overpack. Each of the MPCs has different internals (baskets) to accommodate distinct fuel characteristics. Each MPC is identified by the maximum quantity of fuel assemblies it is capable of receiving. The MPC-24, MPC-24E, and MPC-24EF contain a maximum of 24 PWR fuel assemblies; the MPC-32 and MPC-32F contains a maximum of 32 PWR fuel assemblies; and the MPC-68, MPC-68F, and MPC-68FF contain a maximum of 68 BWR fuel assemblies.

The HI-STORM overpack is constructed from a combination of steel and concrete, both of which are materials with long, proven histories of usage in nuclear applications. The HI-STORM overpack incorporates and combines many desirable features of previously-approved concrete and metal module designs. In essence, the HI-STORM overpack is a hybrid of metal and concrete systems, with the design objective of emulating the best features and dispensing with the drawbacks of both. The HI-STORM overpack is best referred to as a METCON™ (metal/concrete composite) system.



Figures 1.1.1 and 1.1.1A show the HI-STORM 100 with two of its major constituents, the MPC and the storage overpack, in a cut-away view. The MPC, shown partially withdrawn from the storage overpack, is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the confinement boundary for the stored spent nuclear fuel assemblies with respect to 10CFR72 requirements and attendant review considerations. The HI-STORM 100 storage overpack provides mechanical protection, cooling, and radiological shielding for the contained MPC.

In essence, the HI-STORM 100 System is the storage-only counterpart of the HI-STAR 100 System (Docket Numbers 72-1008 (Ref. [1.1.2]) and 71-9261 (Ref. [1.1.3])). Both HI-STORM and HI-STAR are engineered to house identical MPCs. Since the MPC is designed to meet the requirements of both 10CFR71 and 10CFR72 for transportation and storage, respectively, the HI-STORM 100 System allows rapid decommissioning of the ISFSI by simply transferring the loaded MPC's directly into HI-STAR 100 overpacks for off-site transport. This alleviates the additional fuel handling steps required by storage-only casks to unload the cask and repackage the fuel into a suitable transportation cask.

In contrast to the HI-STAR 100 overpack, which provides a containment boundary for the SNF during transport, the HI-STORM storage overpack does not constitute a containment or confinement enclosure. The HI-STORM overpack is equipped with large penetrations near its lower and upper extremities to permit natural circulation of air to provide for the passive cooling of the MPC and the contained radioactive material. The HI-STORM *overpack* is engineered to be an effective barrier against the radiation emitted by the stored materials, and an efficiently configured metal/concrete composite to attenuate the loads transmitted to the MPC during a natural phenomena or hypothetical accident event. Other auxiliary functions of the HI-STORM *100* overpack include isolation of the SNF from abnormal environmental or man-made events, such as impact of a tornado borne missile. As the subsequent chapters of this FSAR demonstrate, the HI-STORM overpack is engineered with large margins of safety with respect to cooling, shielding, and mechanical/structural functions.

The HI-STORM 100 System is autonomous inasmuch as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM 100 System since it is completely passive and is composed of materials with long proven histories in the nuclear industry. The HI-STORM 100 System can be used either singly or as the basic storage module in an ISFSI. The site for an ISFSI can be located either at a reactor or away from a reactor.

The information presented in this report is intended to demonstrate the acceptability of the HI-STORM 100 System for use under the general license provisions of Subpart K by meeting the criteria set forth in 10CFR72.236.

The modularity of the HI-STORM 100 System accrues several advantages. Different MPCs, identical in exterior dimensions, manufacturing requirements, and handling features, but different in their SNF arrangement details, are designed to fit a common overpack. Even though the different MPCs have fundamentally identical design and manufacturing attributes, qualification of HI-STORM 100 requires consideration of the variations in the characteristics of the MPCs. In most

cases, however, it is possible to identify the most limiting MPC geometry and the specific loading condition for the safety evaluation, and the detailed analyses are then carried out for that bounding condition. In those cases where this is not possible, multiple parallel analyses are performed.

The HI-STORM overpack is not engineered for transport and, therefore, will not be submitted for 10CFR Part 71 certification. HI-STORM 100, however, is designed to possess certain key elements of flexibility.

For example:

- The HI-STORM overpack is stored at the ISFSI pad in a vertical orientation which helps minimize the size of the ISFSI and leads to an effective natural convection cooling flow around the MPC.
- The HI-STORM overpack can be loaded with a loaded MPC using the HI-TRAC transfer cask inside the 10CFR50 [1.1.4] facility, prepared for storage, transferred to the ISFSI, and stored in a vertical configuration, or directly loaded using the HI-TRAC transfer cask at or nearby the ISFSI storage pad.

The version of the HI-STORM overpack equipped with sector lugs to anchor it to the ISFSI pad is labeled HI-STORM 100A, shown in Figure 1.1.4. Figure 1.1.5 shows the sector lugs and anchors used to fasten the overpack to the pad in closer view. Details on HI-STORM 100A are presented in the drawing and BOM contained in Section 1.5. Users may employ a double nut arrangement as an option. The HI-STORM 100A overpack will be deployed at those ISFSI sites where the postulated seismic event (defined by the three orthogonal ZPAs) exceeds the maximum limit permitted for free-standing installation. The design of the ISFSI pad and the embedment are necessarily site-specific and the responsibility of the ISFSI owner. These designs shall be in accordance with the requirements specified in Appendix 2.A. The jurisdictional boundary between the anchored cask design and the embedment design is defined in Table 2.0.5. Additional description on the HI-STORM 100A configuration is provided in Subsection 1.2.1.2.1.

The MPC is a multi-purpose SNF storage device both with respect to the type of fuel assemblies and its versatility of use. The MPC is engineered as a cylindrical prismatic structure with square cross section storage cavities. The number of storage locations depends on the type of fuel. Regardless of the storage cell count, the construction of the MPC is fundamentally the same; it is built as a honeycomb of cellular elements positioned within a circumscribing cylindrical canister shell. The manner of cell-to-cell weld-up and cell-to-canister shell interface employed in the MPC imparts extremely high structural stiffness to the assemblage, which is an important attribute for mechanical accident events. Figure 1.1.2 shows an elevation cross section of an MPC.

The MPC is identical to those presented in References [1.1.2] and [1.1.3], except for MPC-24E, 24EF, 32, 32F and 68FF, until such time as those CoCs are amended to include these additional MPC models. Referencing these documents, as applicable, avoids repetition of information on the MPCs which is comprehensively set forth in the above-mentioned Holtec International documents docketed with the NRC. However, sufficient information and drawings are presented in this report to maintain clarity of exposition of technical data.

The HI-STORM storage overpack is designed to provide the necessary neutron and gamma shielding to comply with the provisions of 10CFR72 for dry storage of SNF at an ISFSI. Cross sectional views of the HI-STORM storage overpacks are presented in Figures 1.1.3 and 1.1.3A. A HI-TRAC transfer cask is required for loading of the MPC and movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. The HI-TRAC is engineered to be emplaced with an empty MPC into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC/MPC assembly is designed to preclude intrusion of pool water into the narrow annular space between the HI-TRAC and the MPC while the assembly is submerged in the pool water. The HI-TRAC transfer cask also allows dry loading (or unloading) of SNF into the MPC.

To summarize, the HI-STORM 100 System has been engineered to:

- minimize handling of the SNF;
- provide shielding and physical protection for the MPC;
- permit rapid and unencumbered decommissioning of the ISFSI;
- require minimal ongoing surveillance and maintenance by plant staff;
- minimize dose to operators during loading and handling;
- allow transfer of the loaded MPC to a HI-STAR overpack for transportation.

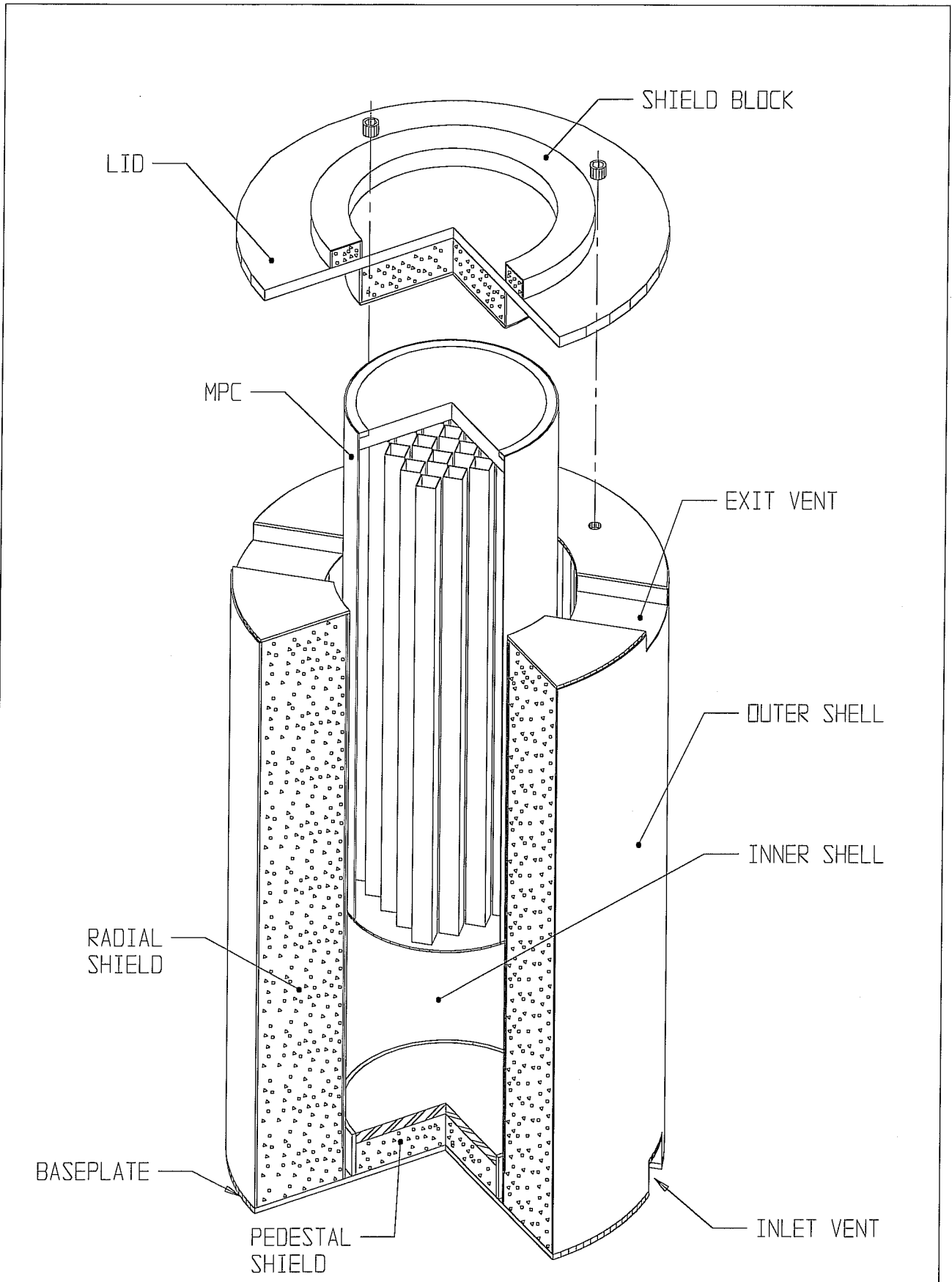


FIGURE 1.1.1; HI-STORM 100 OVERPACK WITH MPC PARTIALLY INSERTED

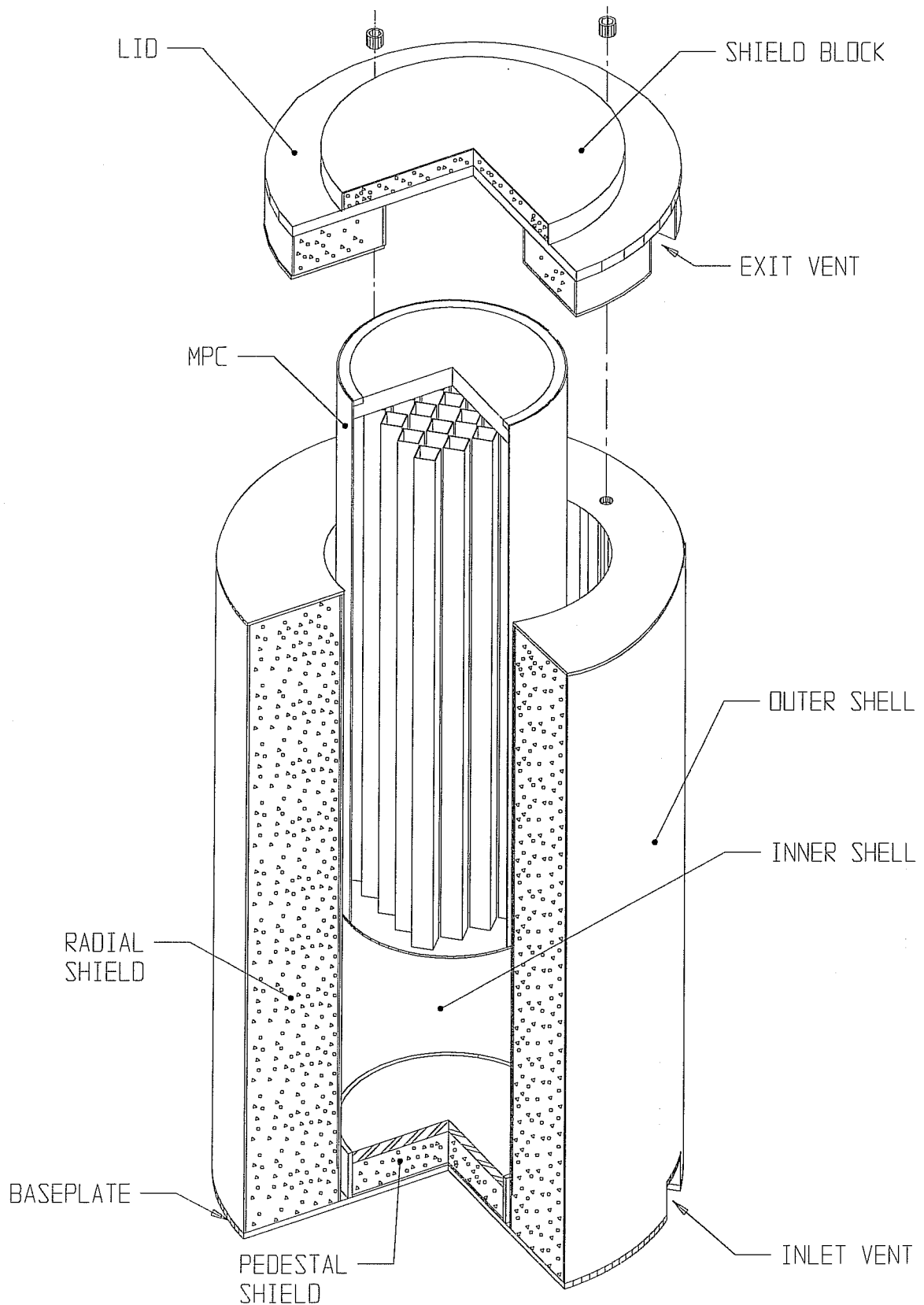


FIGURE 1.1.1A; HI-STORM 100S OVERPACK WITH MPC PARTIALLY INSERTED

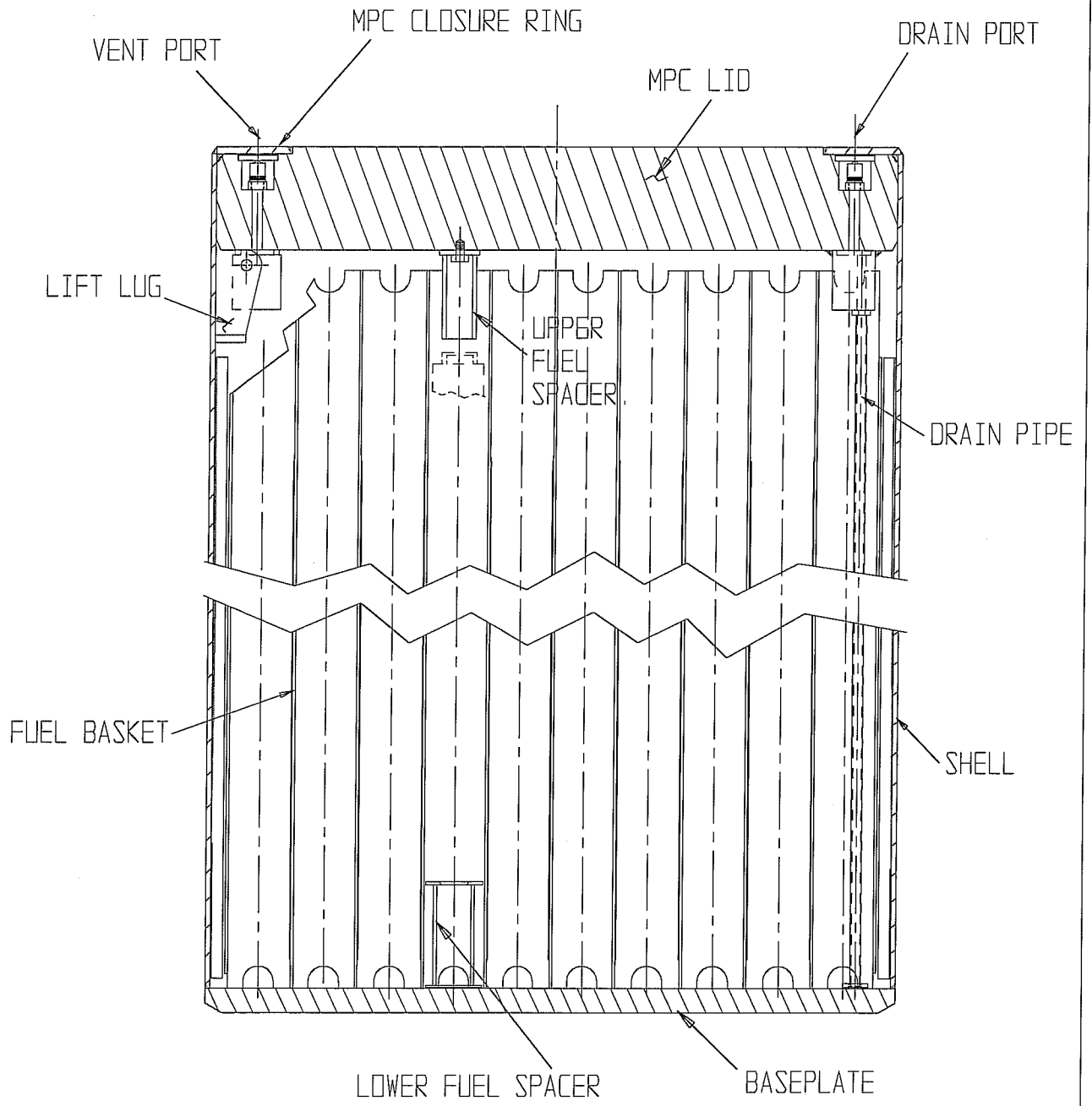


FIGURE 1.1.2; CROSS SECTION ELEVATION VIEW OF MPC

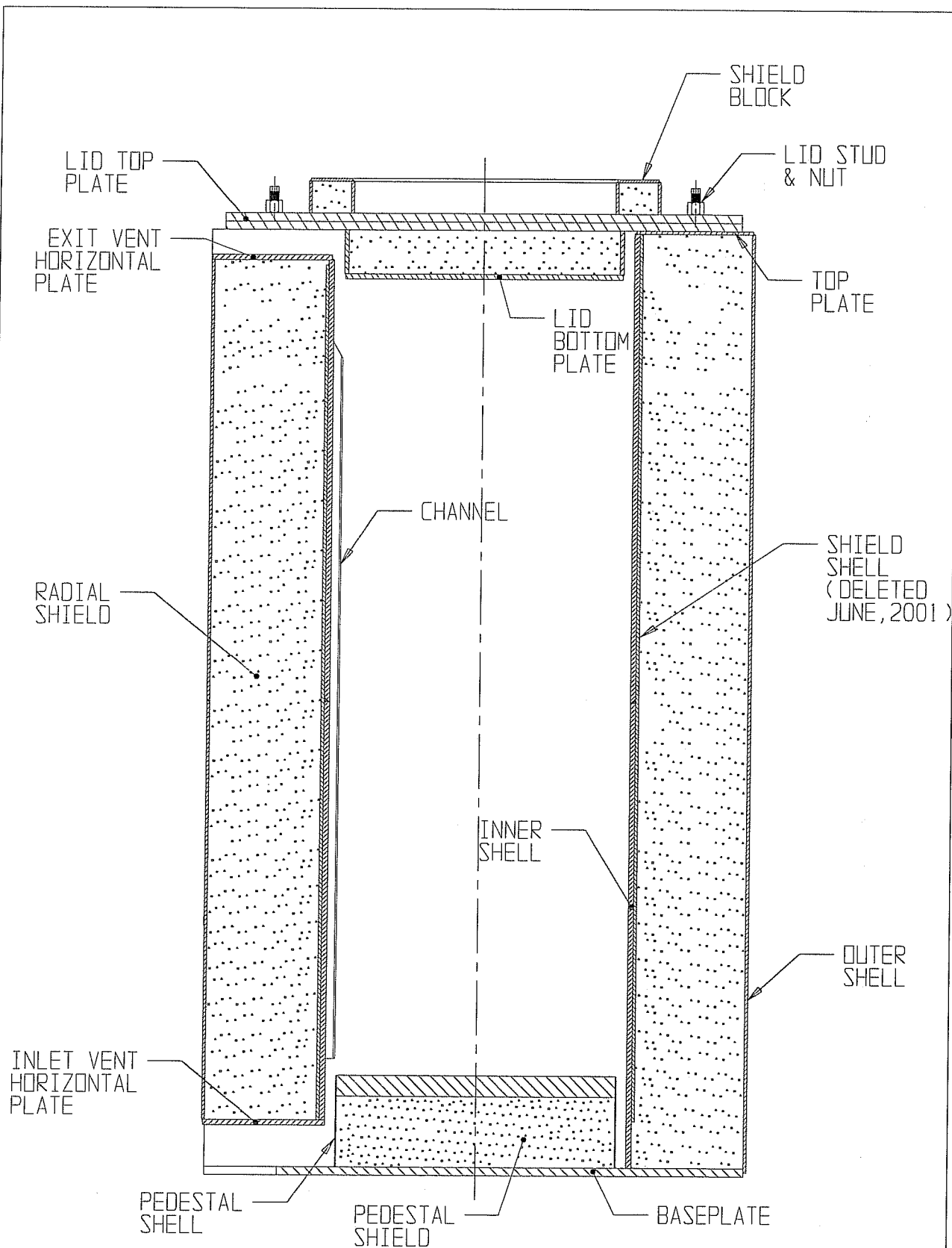


FIGURE 1.1.3; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

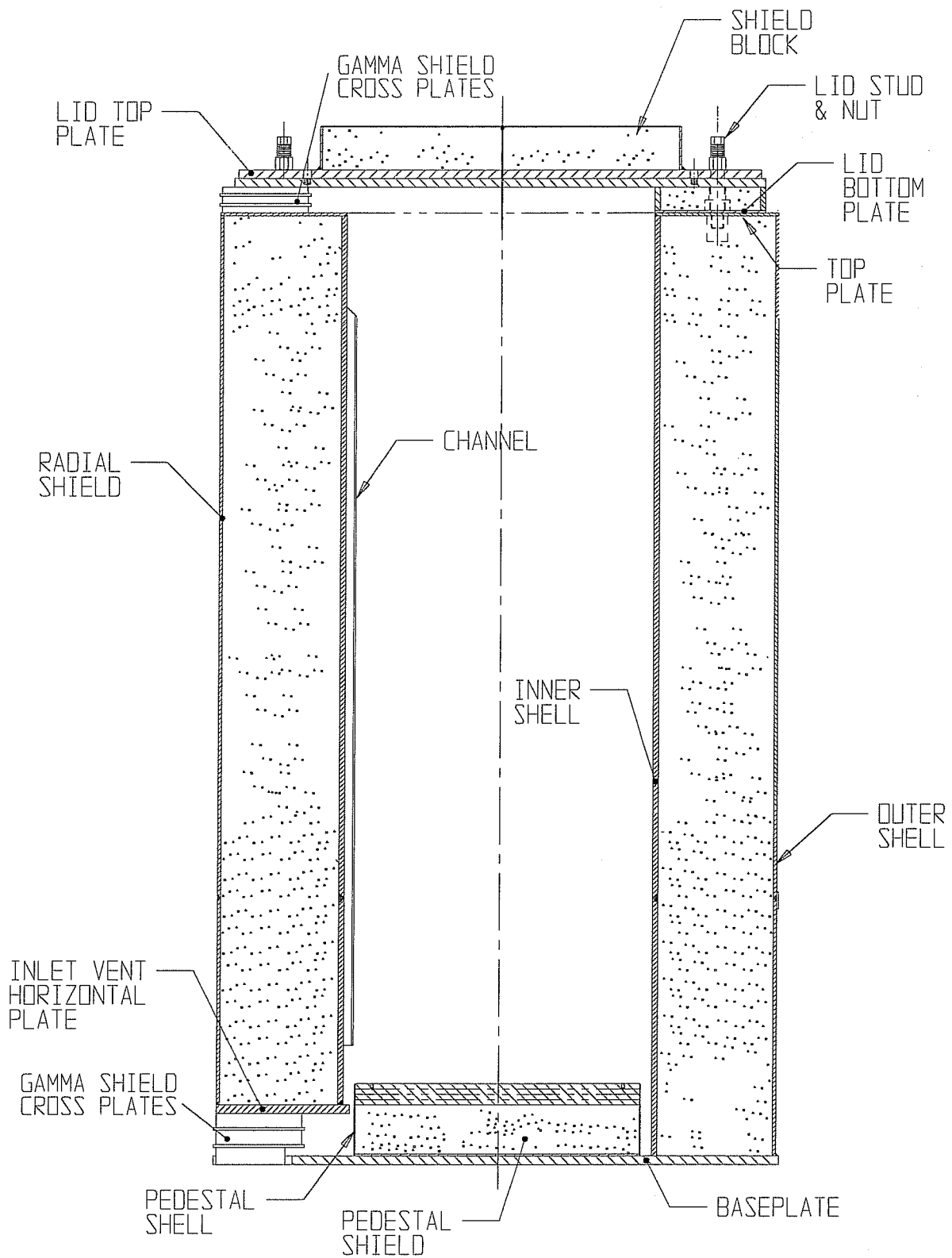
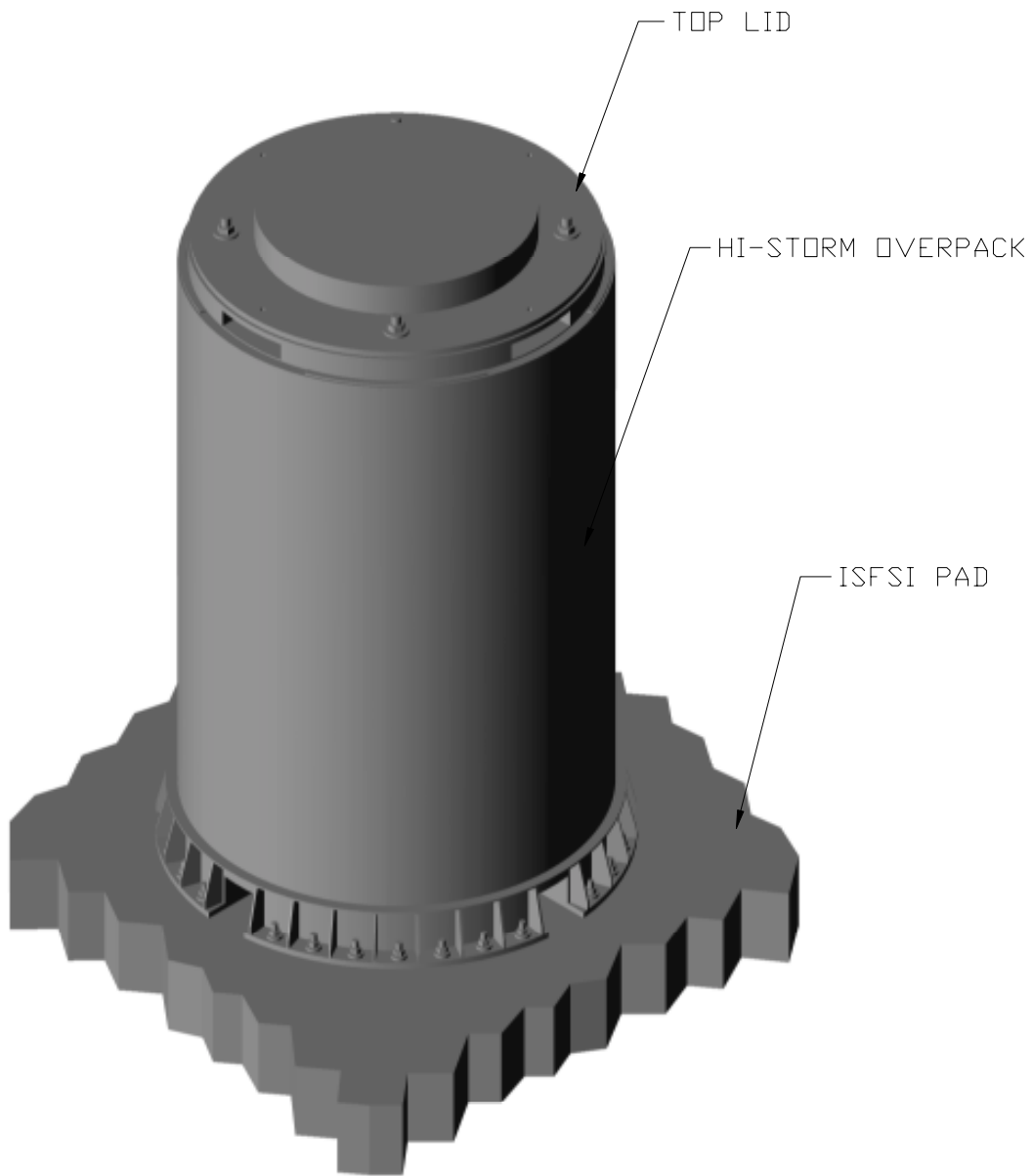
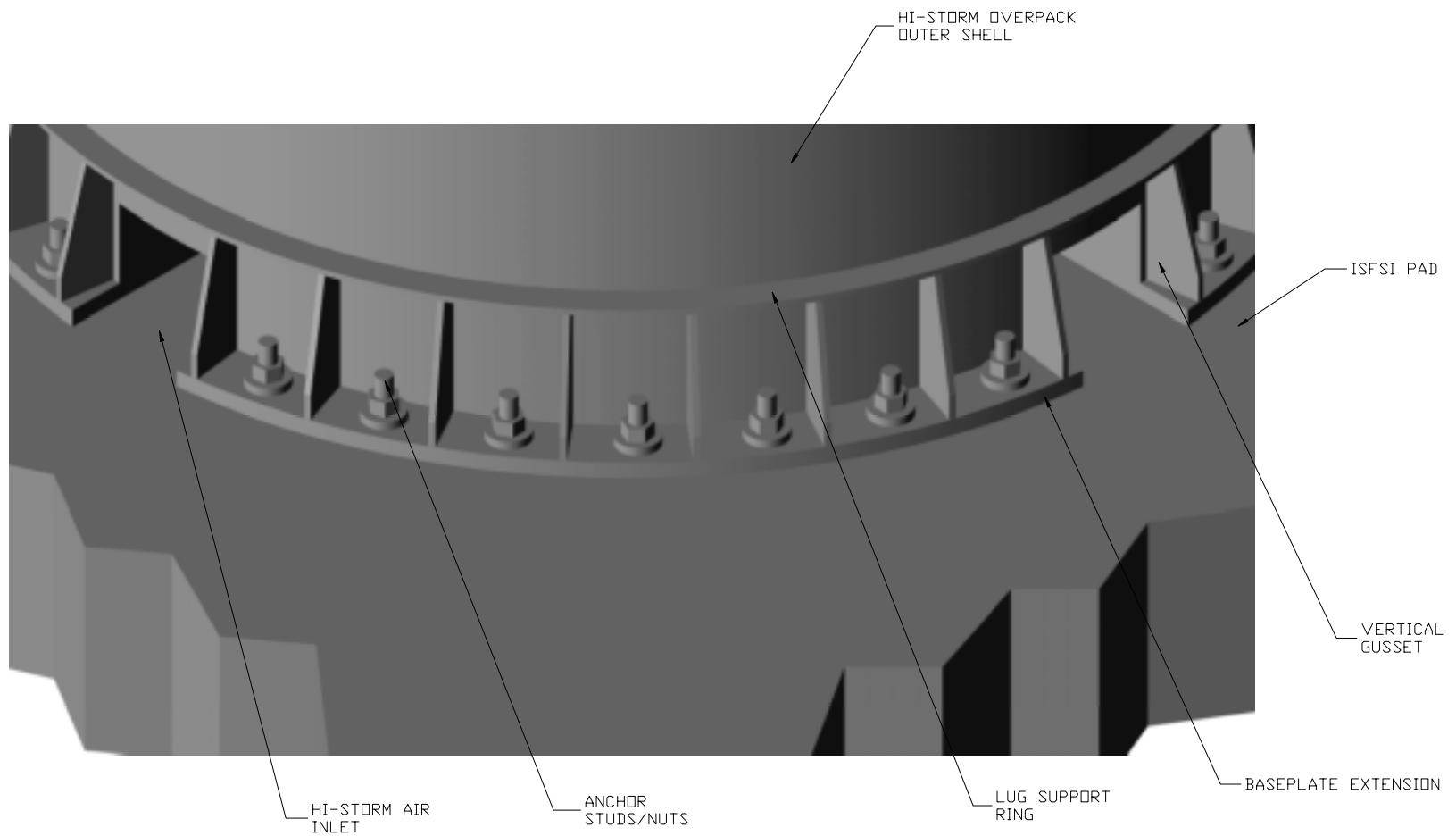


FIGURE 1.1.3A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW





**FIGURE 1.1.4; A PICTORAL VIEW OF THE HI-STORM 100A OVERPACK (100SA MODEL SHOWN)**



**FIGURE 1.1.5; ANCHORING DETAIL FOR THE HI-STORM 100A AND 100SA OVERPACKS**

## 1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

### 1.2.1 System Characteristics

The basic HI-STORM 100 System consists of interchangeable MPCs providing a confinement boundary for BWR or PWR spent nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the storage overpack. Figure 1.2.1 provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a storage overpack. Figure 1.2.1A provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a HI-STORM 100S storage overpack. Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. All structures, systems, and components of the HI-STORM 100 System which are identified as Important to Safety are specified in Table 2.2.6. This discussion is supplemented with a full set of detailed design drawings in Section 1.5.

The HI-STORM 100 System is comprised of three discrete components:

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

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

- i. vacuum drying (or other moisture removal) system
- ii. helium (He) backfill system ~~with leakage detector~~
- iii. lifting and handling systems
- iv. welding equipment
- v. transfer vehicles/trailer

All MPCs have identical exterior dimensions that render them interchangeable. The outer diameter of the MPC is 68-3/8 inches<sup>†</sup> and the overall length is 190-1/2 inches. See Section 1.5 for the ~~detailed design~~ MPC drawings. Due to the differing storage contents of each MPC, the maximum loaded weight differs among MPCs. See Table 3.2.1 for each MPC weight. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons. Tables 1.2.1 and 1.2.2 contain the key *system data and* parameters for the MPCs.

A single, base HI-STORM overpack design is provided which is capable of storing each type of MPC. The overpack inner cavity is sized to accommodate the MPCs. The inner diameter of the overpack inner shell is 73-1/2 inches and the height of the cavity is 191-1/2 inches. The overpack

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<sup>†</sup> Dimensions discussed in this section are considered nominal values.

inner shell is provided with channels distributed around the inner cavity to present an inside diameter of 69-1/2 inches. The channels are intended to offer a flexible medium to absorb some of the impact during a non-mechanistic tip-over, while still allowing the cooling air flow through the ventilated overpack. The outer diameter of the overpack is 132-1/2 inches. The overall height of the HI-STORM 100 and the HI-STORM 100S is 239-1/2 inches. There are two versions of the HI-STORM 100S overpack, differing only in height and weight. The HI-STORM 100S(232) is 232 inches high, and the HI-STORM 100S(243) is 243 inches high. The HI-STORM 100S(243) is approximately 10,100 lbs heavier, including concrete. Hereafter in the text, these two versions of the HI-STORM 100S overpack will only be referred to as HI-STORM 100S and will be discussed separately only if the design feature being discussed is different between the two overpacks. See Section 1.5 for drawings. The weight of the overpack without an MPC is approximately 135 tons. See Table 3.2.1 for the detailed weights.

Before proceeding to present detailed physical data on the HI-STORM 100 System, it is of contextual importance to summarize the design attributes which enhance the performance and safety of the system. Some of the principal features of the HI-STORM 100 System which enhance its effectiveness as an SNF storage device and a safe SNF confinement structure are:

- the honeycomb design of the MPC fuel basket;
- the effective distribution of neutron and gamma shielding materials within the system;
- the high heat dissipation capability;
- engineered features to promote convective heat transfer;
- the structural robustness of the steel-concrete-steel overpack construction.

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

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass of the basket over the entire length of the basket. Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform basket. In other words, the honeycomb basket is a most effective radiation attenuation device. The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the MPC an effective heat rejection device.

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

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

#### 1.2.1.1 Multi-Purpose Canisters

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

There are ~~seven~~ *eight* MPC models, distinguished by the type and number of fuel assemblies authorized for loading. *Section 1.2.3 and Table 1.2.1 summarize the allowable contents for each MPC model. Section 2.1.9 provides the detailed specifications for the contents authorized for storage in the HI-STORM 100 System. The MPC-24 is designed to store up to 24 intact PWR fuel assemblies. The MPC-24E is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies. The MPC-24EF is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies or fuel classified as fuel debris. The MPC-68 is designed to store up to 68 total BWR fuel assemblies including up to 68 damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies. Damaged BWR fuel assemblies other than Dresden Unit 1 and Humboldt Bay are limited to 16 fuel storage locations in the MPC-68 with the remainder being intact BWR fuel assemblies, up to a total of 68. The MPC-68F is designed to store up to 68 intact or damaged Dresden Unit 1 and Humboldt Bay BWR fuel assemblies. Up to four of the 68 fuel storage locations in the MPC-68F may be Dresden Unit 1 and Humboldt Bay BWR fuel assemblies classified as fuel debris. The MPC-68FF is designed to store up to 68 total BWR fuel assemblies including up to 16 damaged BWR fuel assemblies. Up to eight (8) of the 16 BWR damaged fuel assembly storage locations may be filled with BWR fuel classified as fuel debris. In addition, all fuel loading combinations permitted in the MPC-68F are also permitted in the MPC-68FF. Design Drawings for all of the MPCs are provided in Section 1.5.*

The MPC provides the confinement boundary for the stored fuel. Figure 1.2.6 provides an elevation view of the MPC confinement boundary. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring. The confinement boundary is a strength-welded enclosure of all stainless steel construction.

The PWR MPC-24, MPC-24E and MPC-24EF differ in construction from the MPC-32 (*including the MPC-32F*) and the MPC-68 (*including the MPC-68F and MPC-68FF*) in one important aspect: the fuel storage cells *in the MPC-24 series* are physically separated from one another by a "flux trap", for criticality control. The PWR MPC-32 *and -32F are is*-designed similar to the MPC-68 (without flux traps) and its design includes credit for soluble boron in the MPC water during wet fuel loading and unloading operations for criticality control.

The MPC fuel baskets of non-flux trap construction (namely, MPC-68, MPC-68F, MPC-68FF, and MPC-32, and MPC-32F) are formed from an array of plates welded to each other at their intersections. In the flux-trap type fuel baskets (MPC-24, MPC-24E, and MPC-24EF), formed angles are interposed onto the orthogonally configured plate assemblage to create the required flux-trap channels (see MPC-24 and MPC-24E ~~design~~-fuel basket drawings in Section 1.5). In both configurations, two key attributes of the basket are preserved:

- i. The cross section of the fuel basket simulates a multi-flanged closed section beam, resulting in extremely high bending rigidity.
- ii. The principal structural frame of the basket consists of co-planar plate-type members (i.e., no offset).

This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls that must transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (e.g., non-mechanistic tipover, uncontrolled lowering of a cask during on-site transfer, or off-site transport events, etc.).

The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. Between the periphery of the basket, the MPC shell, and the basket supports, optional *aluminum* heat conduction elements (*AHCEs*) may have ~~be~~ been installed in the early vintage MPCs fabricated, certified, and loaded under the original version or Amendment 1 of the HI-STORM 100 System CoC. The presence of these aluminum heat conduction elements is acceptable for MPCs loaded under the original CoC or Amendment 1, since the governing thermal analysis for Amendment 1 conservatively modeled the AHCEs as restrictions to convective flow in the basket, but took no credit for heat transfer through them. The heat loads authorized under Amendment 1 bound those for the original CoC, with the same MPC design. For MPCs loaded under Amendment 2 or a later version of the HI-STORM 100 CoC, the aluminum heat conduction elements shall not be installed. MPCs both with and without aluminum heat conduction elements installed are compatible with all HI-STORM overpacks. If used, ~~these~~ these heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design that allows a snug fit in the confined spaces and ease of installation. If used, the heat conduction elements are installed along the full length of the MPC basket except at the drain pipe location to create a nonstructural thermal connection that facilitates heat transfer from the basket to shell. In their operating condition, the heat conduction elements contact the MPC shell and basket walls.

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

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

To maintain a constant exterior axial length between the PWR MPCs and the BWR MPCs the thickness of the PWR MPCs' lid is ½ inch thinner than the MPC-68 's-2 lid to accommodate the longest PWR fuel assembly which is approximately a ½ inch longer than the longest BWR fuel assembly. For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested values for the upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type. The actual length of fuel spacers will be determined on a site-specific or fuel assembly-specific basis.

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

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

At this time, there is considerable uncertainty with respect to the material of construction for an MPC that would be acceptable as a waste package for the MGDS. Candidate materials being considered for acceptability by the DOE include:

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

The DOE material selection process is primarily driven by corrosion resistance in the potential environment of the MGDS. As the decision regarding a suitable material to meet disposal requirements is not imminent, the MPC design allows the use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this FSAR) may be one of the following materials. Only a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC basket or shell - the basket and shell may be of different alloys in the same MPC.

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

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

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

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

### 1.2.1.2 Overpacks

#### 1.2.1.2.1 HI-STORM 100 Overpack (Storage)

The HI-STORM 100 and 100S overpacks are rugged, heavy-walled cylindrical vessels. Figures 1.2.7, 1.2.8, and 1.2.8A provide cross sectional views of the HI-STORM 100 System, showing both of the overpack designs, respectively. The HI-STORM 100A is an anchored variant of the same structure and hereinafter is identified by name only when the discussion specifically applies to the anchored overpack. The HI-STORM 100A differs only in the diameter of the overpack baseplate and the presence of bolt holes and associated anchorage hardware (see Figures 1.1.4 and 1.1.5). The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The overpack plain concrete is enclosed by cylindrical steel shells, a thick steel baseplate, and a top plate. The



overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation in the vertical direction.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC. The inner shell of the overpack has channels attached to its inner diameter. The channels provide guidance for MPC insertion and removal and a flexible medium to absorb impact loads during the non-mechanistic tip-over, while still allowing the cooling air flow to circulate through the overpack. Shims may be attached to channels to allow the proper inner diameter dimension to be obtained.

The storage system has air ducts to allow for passive natural convection cooling of the contained MPC. A *minimum of* ~~four~~ air inlets and four air outlets are located at the lower and upper extremities of the storage system, respectively. The location of the air outlets in the HI-STORM 100 and the HI-STORM 100S design differ in that the outlet ducts for the HI-STORM 100 overpack are located in the overpack body and are aligned vertically with the inlet ducts at the bottom of the overpack body. The air outlet ducts in the HI-STORM 100S are integral to the lid assembly and are not in vertical alignment with the inlet ducts. The location of the air inlet ducts is same for both the HI-STORM 100 and the HI-STORM 100S. The air inlets and outlets are covered by a fine mesh screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, described in Chapter 11 of this FSAR, evaluates the effects of partial and complete blockage of the air ducts.

The ~~four~~ air inlets and ~~four~~ air outlets are penetrations through the thick concrete shielding provided by the HI-STORM 100 overpack. The outlet air ducts for the HI-STORM 100S overpack, integral to the lid, present a similar break in radial shielding. Within the air inlets and outlets, an array of gamma shield cross plates are installed (see Figure 5.3.19 for a pictorial representation of the gamma shield cross plate designs). These gamma shield cross plates are designed to scatter any particles traveling through the ducts. The result of scattering the particles in the ducts is a significant decrease in the local dose rates around the four air inlets and four air outlets. The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized. The shielding analysis conservatively credits only the mandatory version of the gamma shield cross plate design because they provide less shielding than the optional design. Conversely, the thermal analysis conservatively evaluates the optional gamma shield cross plate design because it conservatively provides greater resistance to flow than the mandatory design.

Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM 100) or the inlet air duct horizontal plates (HI-STORM 100S) (see Figure 1.2.7). The four anchor blocks are located on 90° arcs around the circumference of the overpack. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, Hillman rollers, or other design based on site-specific needs and capabilities. Slings or other suitable

devices mate with lifting lugs that are inserted into threaded holes in the top surface of the overpack lid to allow lifting of the overpack lid. After the lid is bolted to the storage overpack main body, these lifting bolts shall be removed and replaced with flush plugs.

The plain concrete between the overpack inner and outer steel shells is specified to provide the necessary shielding properties (dry density) and compressive strength. The concrete shall be in accordance with the requirements specified in Appendix 1.D.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, in an implicit manner it helps enhance the performance of the HI-STORM overpack in other respects as well. For example, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. The case of a postulated fire accident at the ISFSI is another example where the high thermal inertia characteristics of the HI-STORM concrete control the temperature of the MPC. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the inter-shell space, such that, while its cracking and crushing under a tip-over accident is not of significant consequence, its deformation characteristics are germane to the analysis of the structural members.

Density and compressive strength are the key parameters which delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal, and HI-STORM lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.0.5] are used.

To ensure the stability of the concrete at temperature, the concrete composition has been specified in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" [1.0.3]. Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM overpack concrete.

There are two base HI-STORM overpack designs - HI-STORM 100 and HI-STORM 100S. The significant differences between the two are overpack height, MPC pedestal height, location of the air outlet ducts, and the vertical alignment of the inlet and outlet air ducts. The HI-STORM 100 overpack is approximately 240 inches high from the bottom of the baseplate to the top of the lid bolts and 227 inches high without the lid installed. There are two versions of the HI-STORM 100S overpack design, differing only in height and weight. The HI-STORM 100S(232) is approximately 232 inches from the bottom of the baseplate to the top of the lid bolts in its final storage configuration and 211 inches high without the lid installed. The HI-STORM 100S(243) is approximately 243 inches from the bottom of the baseplate to the top of the lid bolts in its final storage configuration and 222 inches high without the lid installed.

The anchored embodiment of the HI-STORM overpack is referred to as HI-STORM 100A. As explained in the foregoing, the HI-STORM overpack is a steel weldment, which makes it a relatively simple matter to extend the overpack baseplate, form lugs, and then anchor the cask to the reinforced concrete structure of the ISFSI. In HI-STORM terminology, these lugs are referred to as “sector lugs.” The sector lugs, as shown in Figure 1.1.5 and the drawing in Section 1.5, are formed by extending the HI-STORM overpack baseplate, welding vertical gussets to the baseplate extension and to the overpack outer shell and, finally, welding a horizontal lug support ring in the form of an annular sector to the vertical gussets and to the outer shell. The baseplate is equipped with regularly spaced clearance holes (round or slotted) through which the anchor studs can pass. The sector lugs are bolted to the ISFSI pad using anchor studs that are made of a creep-resistant, high-ductility, environmentally compatible material. The bolts are pre-loaded to a precise axial stress using a “stud tensioner” rather than a torque wrench. Pre-tensioning the anchors using a stud tensioner eliminates any shear stress in the bolt, which is unavoidable if a torquing device is employed (Chapter 3 of the text “Mechanical Design of Heat Exchangers and Pressure Vessel Components”, by Arcturus Publishers, 1984, K.P. Singh and A.I. Soler, provides additional information on stud tensioners). The axial stress in the anchors induced by pre-tensioning is kept below 75% of the material yield stress, such that during the seismic event the maximum bolt axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF (for Level D conditions). Figures 1.1.4 and 1.1.5 provide visual depictions of the anchored HI-STORM 100A configuration. This configuration also applies to the HI-STORM 100SA.

The anchor studs pass through liberal clearance holes (circular or slotted) in the sector lugs (0.75” minimum clearance) such that the fastening of the studs to the ISFSI pad can be carried out without mechanical interference from the body of the sector lug. The two clearance hole configurations give the ISFSI pad designer flexibility in the design of the anchor embedment in the ISFSI concrete. The axial force in the anchors produces a compressive load at the overpack/pad interface. This compressive force,  $F$ , imparts a lateral load bearing capacity to the cask/pad interface that is equal to  $\mu F$  ( $\mu \leq 0.53$  per Table 2.2.8). As is shown in Chapter 3 of this FSAR, the lateral load-bearing capacity of the HI-STORM/pad interface ( $\mu F$ ) is many times greater than the horizontal (sliding) force exerted on the cask under the postulated DBE seismic event. Thus, the potential for lateral sliding of the HI-STORM 100A System during a seismic event is precluded, as is the potential for any bending action on the anchor studs.

The seismic loads, however, will produce an overturning moment on the overpack that would cause a redistribution of the compressive contact pressure between the pad and the overpack. To determine the pulsation in the tensile load in the anchor studs and in the interface contact pressure, bounding static analysis of the preloaded configuration has been performed. The results of the static analysis demonstrate that the initial preloading minimizes pulsations in the stud load. A confirmatory non-linear dynamic analysis has also been performed using the time-history methodology described in Chapter 3, wherein the principal nonlinearities in the cask system are incorporated and addressed. The calculated results from the dynamic analysis confirm the static analysis results and that the presence of pre-stress helps minimize the pulsation in the anchor stud stress levels during the seismic event, thus eliminating any concern with regard to fatigue failure under extended and repetitive seismic excitations.

The sector lugs in HI-STORM 100A are made of the same steel material as the baseplate and the shell (SA516- Gr. 70) which helps ensure high quality fillet welds used to join the lugs to the body of the overpack. The material for the anchor studs can be selected from a family of allowable stud materials listed in the ASME Code (Section II). A representative sampling of permitted materials is listed in Table 1.2.7. The menu of materials will enable the ISFSI owner to select a fastener material that is resistant to corrosion in the local ISFSI environment. For example, for ISFSIs located in marine environments (e.g., coastal reactor sites), carbon steel studs would not be recommended without concomitant periodic inspection and coating maintenance programs. Table 1.2.7 provides the chemical composition of several acceptable fastener materials to help the ISFSI owner select the most appropriate material for his site. The two mechanical properties, ultimate strength  $\sigma_u$  and yield strength  $\sigma_y$  are also listed. For purposes of structural evaluations, the lower bound values of  $\sigma_u$  and  $\sigma_y$  from the menu of materials listed in Table 1.2.7 are used (see Table 3.4.10).

As shown in the drawing, the anchor studs are spaced sufficiently far apart such that a practical reinforced concrete pad with embedded receptacles can be designed to carry the axial pull from the anchor studs without overstressing the enveloping concrete monolith. The design specification and supporting analyses in this FSAR are focused on qualifying the overpack structures, including the sector lugs and the anchor studs. The design of the ISFSI pad, and its anchor receptacle will vary from site to site, depending on the geology and seismological characteristics of the sub-terrain underlying the ISFSI pad region. The data provided in this FSAR, however, provide the complete set of factored loads to which the ISFSI pad, its sub-grade, and the anchor receptacles must be designed within the purview of ACI-349-97 [1.0.4]. Detailed requirements on the ISFSI pads for anchored casks are provided in Section 2.0.4.

#### 1.2.1.2.2 HI-TRAC (Transfer Cask) - Standard Design

Like the storage overpack, the HI-TRAC transfer cask is a rugged, heavy-walled cylindrical vessel. The main structural function of the transfer cask is provided by carbon steel, and the main neutron and gamma shielding functions are provided by water and lead, respectively. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. Figure 1.2.9 provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. See Section 1.2.1.2.3 for discussion of the optional HI-TRAC 125D design.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC. The top lid of the HI-TRAC 125 has additional neutron shielding to provide neutron attenuation in the vertical direction (from SNF in the MPC below). The MPC access hole through the HI-TRAC top lid is provided to allow the lowering/raising of the MPC between the HI-TRAC transfer cask, and the HI-STORM or HI-STAR overpacks. The standard design HI-TRAC (comprised of HI-TRAC 100 and HI-TRAC 125) is provided with two bottom lids, each used separately. The pool lid is bolted to the bottom flange of the HI-TRAC and is utilized during MPC fuel loading and sealing operations. In addition to providing shielding in the axial direction, the pool lid incorporates a seal that is designed to hold clean demineralized water in the HI-TRAC inner cavity, thereby preventing contamination of the exterior of the MPC by the contaminated fuel pool water. After the MPC has been drained, dried, and sealed, the pool lid is removed and the HI-TRAC transfer lid is attached (standard design only). The transfer lid

incorporates two sliding doors that allow the opening of the HI-TRAC bottom for the MPC to be raised/lowered. Figure 1.2.10 provides a cross section of the HI-TRAC with the transfer lid installed.

In the standard design, trunnions are provided for lifting and rotating the transfer cask body between vertical and horizontal positions. The lifting trunnions are located just below the top flange and the pocket trunnions are located above the bottom flange. The two lifting trunnions are provided to lift and vertically handle the HI-TRAC, and the pocket trunnions provide a pivot point for the rotation of the HI-TRAC for downending or upending.

Two standard design HI-TRAC transfer casks of different weights are provided to house the MPCs. The 125 ton HI-TRAC weight does not exceed 125 tons during any loading or transfer operation. The 100 ton HI-TRAC weight does not exceed 100 tons during any loading or transfer operation. The internal cylindrical cavities of the two standard design HI-TRACs are identical. However, the external dimensions are different. The 100ton HI-TRAC has a reduced thickness of lead and water shielding and consequently, the external dimensions are different. The structural steel thickness is identical in the two HI-TRACs. This allows most structural analyses of the 125 ton HI-TRAC to bound the 100 ton HI-TRAC design. Additionally, as the two HI-TRACs are identical except for a reduced thickness of lead and water, the 125 ton HI-TRAC has a larger thermal resistance than the smaller and lighter 100 ton HI-TRAC. Therefore, for normal conditions the 125 ton HI-TRAC thermal analysis bounds that of the 100 ton HI-TRAC. Separate shielding analyses are performed for each HI-TRAC since the shielding thicknesses are different between the two.

#### 1.2.1.2.3 HI-TRAC 125D Transfer Cask

As an option to using either of the standard HI-TRAC transfer cask design, users may choose to use the optional HI-TRAC 125D design. Figure 1.2.9A provides a typical cross section of the standard design HI-TRAC-125 with the pool lid installed. Like the standard design, the HI-TRAC 125D is designed and constructed in accordance with ASME III, Subsection NF, with certain NRC-approved alternatives, as discussed in Section 2.2.4. Functionally equivalent, the major differences between the HI-TRAC 125D design and the standard design are as follows:

- No pocket trunnions are provided for downending/upending
- The transfer lid is not required
- A new ancillary, the HI-STORM mating device (Figure 1.2.18) is required during MPC transfer operations
- A wider baseplate with attachment points for the mating device is provided
- The baseplate incorporates gussets for added structural strength
- The number of pool lid bolts is reduced

The interface between the MPC and the transfer cask is the same between the standard design and the HI-TRAC 125D design. The optional design is capable of withstanding all loads defined in the design basis for the transfer cask during normal, off-normal, and accident modes of

operation with adequate safety margins. In lieu of swapping the pool lid for the transfer lid to facilitate MPC transfer, the pool lid remains on the HI-TRAC 125D until MPC transfer is required. The HI-STORM mating device is located between, and secured with bolting to, the top of the HI-STORM overpack and the HI-TRAC 125D transfer cask. The mating device is used to remove the pool lid to provide a pathway for MPC transfer between the overpack and the transfer cask. Section 1.2.2.2 provides additional detail on the differences between the standard transfer cask design and the HI-TRAC 125D design during operations.

### 1.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure the radiation and exposure requirements in 10CFR72.104 and 10CFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the SNF in the MPC for ALARA considerations during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and Boral neutron poison absorber panels attached to the fuel storage cell vertical surfaces provide the initial attenuation of gamma and neutron radiation emitted by the radioactive spent fuel. The MPC shell, baseplate, lid and closure ring provide additional thicknesses of steel to further reduce the gamma flux at the outer canister surfaces.

In the HI-STORM storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the lid, and a thick circular concrete pedestal upon which the MPC rests. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shielding integral to the lid provide additional gamma attenuation in the upward direction, reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness. Gamma shield cross plates, as depicted in Figure 5.3.19, provide attenuation of scattered gamma radiation as it exits the inlet and outlet air ducts.

In the HI-TRAC transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the axial direction, shielding is provided by the top lid, and the pool or transfer lid, as applicable. In the HI-TRAC pool lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC. In the transfer lid, layers of steel-lead-steel provide gamma attenuation. For the HI-TRAC 125 transfer lid, the neutron shield material, Holtite-A, is also provided. The HI-TRAC 125 and HI-TRAC 125D top lids are composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The HI-TRAC 100 top lid is composed of steel only providing gamma attenuation.

### 1.2.1.3.1 Boral-Fixed Neutron Absorbers

#### 1.2.1.3.1.1 *Boral*<sup>TM</sup>

Boral is a thermal neutron poison material composed of boron carbide and aluminum (aluminum powder and plate). Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The Boral cladding is made of alloy aluminum, a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal, and chemical environment of a nuclear reactor, spent fuel pool, or dry cask. See Section 3.4.1 for discussion of the reaction of Boral with spent fuel pool water during fuel loading and unloading operations.

The documented historical applications of Boral, in environments comparable to those in spent fuel pools and fuel storage casks, dates to the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor [1.2.2]). Technical data on the material was first printed in 1949, when the report "Boral: A New Thermal Neutron Shield" was published [1.2.3]. In 1956, the first edition of the Reactor Shielding Design Manual [1.2.4] was published and it contained a section on Boral and its properties.

In the research and test reactors built during the 1950s and 1960s, Boral was frequently the material of choice for control blades, thermal-column shutters, and other items requiring very good thermal-neutron absorption properties. It is in these reactors that Boral has seen its longest service in environments comparable to today's applications.

Boral found other uses in the 1960s, one of which was a neutron poison material in baskets used in the shipment of irradiated, enriched fuel rods from Canada's Chalk River laboratories to Savannah River. Use of Boral in shipping containers continues, with Boral serving as the poison in current British Nuclear Fuels Limited casks and the Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

Boral has been exclusively used in fuel storage applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and several unique characteristics, such as:

- The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.

- Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- Boral is stable, strong, durable, and corrosion resistant.

Boral absorbs thermal neutrons without physical change or degradation of any sort from the anticipated exposure to gamma radiation and heat. The material does not suffer loss of neutron attenuation capability when exposed to high levels of radiation dose.

Holtec International's QA Program ensures that Boral is manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR72, Subpart G. Holtec International has procured over 200,000 panels of Boral from AAR Advanced Structures in over 30 projects. Boral has always been purchased with a minimum  $^{10}\text{B}$  loading requirement. Coupons extracted from production runs were tested using the wet chemistry procedure. The actual  $^{10}\text{B}$  loading, out of thousands of coupons tested, has never been found to fall below the design specification. The size of this coupon database is sufficient to provide reasonable assurance that all future Boral procurements will continue to yield Boral with full compliance with the stipulated minimum loading. Furthermore, the surveillance, coupon testing, and material tracking processes which have so effectively controlled the quality of Boral are expected to continue to yield Boral of similar quality in the future. Nevertheless, to add another layer of insurance, only 75%  $^{10}\text{B}$  credit of the fixed neutron absorber is assumed in the criticality analysis consistent with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

The oxide layer that is created from the reaction of the outer aluminum cladding and the edges of the Boral panels with air and water provides a barrier to further reaction of the aluminum cladding with air or the spent fuel pool water during loading and unloading operations. However, with extended submergence in an MPC filled with water or in the plant's spent fuel pool, the hydrodynamic pressure can drive water into the Boral core (comprised of particulate  $\text{B}_4\text{C}$  and aluminum powder) where previously unexposed aluminum powder may react with the water to create hydrogen. The rate of hydrogen generation and the total hydrogen generated is dependent on several variables:

- Aluminum particle size: Aluminum particle size in the Boral core and associated porosity affects the amount of aluminum available for reaction with water. Larger aluminum particles yield less surface area for reaction, but higher porosity for aluminum-water interaction; smaller aluminum particles yield more surface area for reaction, but lower porosity for aluminum-water reaction.



- Presence of trace impurities: The presence of trace impurities in the Boral core due to the manufacturing process (i.e., sodium hydroxide, boron oxide, and iron-oxide) can affect the rate of hydrogen production, both increasing and suppressing the reaction. Sodium dissolved in the water increases the pH and tends to increase the rate of hydrogen production. This is counteracted by the boron oxide, which hydrolyzes to boric acid ( $H_3BO_3$ ) and reduces the rate of hydrogen production. Trace impurities do not affect the total amount of hydrogen generated.
- Pool water chemistry: Chemicals in the plant spent fuel pool water (e.g., copper, boron) can affect the rate of hydrogen production, both increasing (copper) and suppressing (boron) the reaction.
- MPC loading operations: Operating needs or preferences by individual utilities as to when, and for how long the MPC is kept at varying water depths in the spent fuel pool, and how long the MPC is kept filled with water outside the spent fuel pool can affect the amount of aluminum in the Boral core that may be exposed to water.

Due to the variability in hydrogen generation from the Boral-water reaction, the operating procedures in Chapter 8 require monitoring for combustible gases and either exhausting or purging the space beneath the MPC lid during loading and unloading operations when an ignition event could occur (i.e., when the space beneath the MPC lid is open to the welding or cutting operation).

#### 1.2.1.3.1.2 METAMIC<sup>®</sup>

*METAMIC<sup>®</sup> is a neutron absorber material developed by the Reynolds Aluminum Company in the mid-1990s for spent fuel reactivity control in dry and wet storage applications. Metallurgically, METAMIC<sup>®</sup> is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy reinforced with Type 1 ASTM C-750 boron carbide. METAMIC<sup>®</sup> is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. Typically, the average  $B_4C$  particle size is between 10 and 15 microns. As described in the U.S. patents held by METAMIC, Inc.<sup>\*†</sup>, the high performance and reliability of METAMIC<sup>®</sup> derives from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields excellent and uniform homogeneity.*

*The powders are carefully blended without binders or other additives that could potentially adversely influence performance. The maximum percentage of  $B_4C$  that can be dispersed in the aluminum alloy 6061 matrix is approximately 40 wt.%, although extensive manufacturing and testing experience is limited to approximately 31 wt.%. The blend of powders is isostatically compacted into a green billet under high pressure and vacuum sintered to near theoretical density. According to the manufacturer, billets of any size can be produced using this technology. The billet is subsequently extruded into one of a number of product forms, ranging from sheet and plate to*

\* U.S. Patent No. 5,965,829, "Radiation Absorbing Refractory Composition".

† U.S. Patent No. 6,042,779, "Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super, Hypereutectic Al/Si."

angle, channel, round and square tube, and other profiles. For the METAMIC<sup>®</sup> sheets used in the MPCs, the extruded form is rolled down into the required thickness.

METAMIC<sup>®</sup> has been subjected to an extensive array of tests sponsored by the Electric Power Research Institute (EPRI) that evaluated the functional performance of the material at elevated temperatures (up to 900°F) and radiation levels (1E+11 rads gamma). The results of the tests documented in an EPRI report (Ref. [1.2.11]) indicate that METAMIC<sup>®</sup> maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state. The main conclusions provided in the above-referenced EPRI report are summarized below:

- The metal matrix configuration produced by the powder metallurgy process with a complete absence of open porosity in METAMIC<sup>®</sup> ensures that its density is essentially equal to the theoretical density.
- The physical and neutronic properties of METAMIC<sup>®</sup> are essentially unaltered under exposure to elevated temperatures (750° F - 900° F).
- No detectable change in the neutron attenuation characteristics under accelerated corrosion test conditions has been observed.

In addition, independent measurements of boron carbide particle distribution show extremely small particle-to-particle distance<sup>†</sup> and near-perfect homogeneity.

An evaluation of the manufacturing technology underlying METAMIC<sup>®</sup> as disclosed in the above-referenced patents and of the extensive third-party tests carried out under the auspices of EPRI makes METAMIC<sup>®</sup> an acceptable neutron absorber material for use in the MPCs. Holtec's technical position on METAMIC<sup>®</sup> is also supported by the evaluation carried out by other organizations (see, for example, USNRC's SER on NUHOMS-61BT, Docket No. 72-1004).

Consistent with its role in reactivity control, all METAMIC<sup>®</sup> material procured for use in the Holtec MPCs will be qualified as important-to-safety (ITS) Category A item. ITS category A manufactured items, as required by Holtec's NRC-approved Quality Assurance program, must be produced to essentially preclude the potential of an error in the procurement of constituent materials and the manufacturing processes. Accordingly, material and manufacturing control processes must be established to eliminate the incidence of errors, and inspection steps must be implemented to serve as an independent set of barriers to ensure that all critical characteristics defined for the material by the cask designer are met in the manufactured product.

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<sup>†</sup> Medium measured neighbor-to-neighbor distance is 10.08 microns according to the article, "METAMIC Neutron Shielding", by K. Anderson, T. Haynes, and R. Kazmier, EPRI Boraflex Conference, November 19-20, 1998.

All manufacturing and in-process steps in the production of METAMIC<sup>®</sup> shall be carried out using written procedures. As required by the company's quality program, the material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances as set forth in the applicable Holtec QA procedures to ensure that all METAMIC<sup>®</sup> panels procured meet with the requirements appropriate for the quality genre of the MPCs. Additional details pertaining to the qualification and production tests for METAMIC<sup>®</sup> are summarized in Subsection 9.1.5.3.

Because of the absence of interconnected porosities, the time required to dehydrate a METAMIC<sup>®</sup>-equipped MPC is expected to be less compared to an MPC containing Boral.

NUREG/CR-5661 (Ref. [1.2.14]) recommends limiting poison material credit to 75% of the minimum <sup>10</sup>B loading because of concerns for potential "streaming" of neutrons, and allows for greater percentage credit in criticality analysis "if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented". The value of 75% is characterized in NUREG/CR-5661 as a very conservative value, based on experiments with neutron poison containing relatively large B<sub>4</sub>C particles, such as BORAL with an average particle size in excess of 100 microns. METAMIC<sup>®</sup>, however, has a much smaller particle size of typically between 10 and 15 microns on average. Any streaming concerns would therefore be drastically reduced.

Analyses performed by Holtec International show that the streaming due to particle size is practically non-existent in METAMIC<sup>®</sup>. Further, EPRI's neutron attenuation measurements on 31 and 15 B<sub>4</sub>C weight percent METAMIC<sup>®</sup> showed that METAMIC<sup>®</sup> exhibits very uniform <sup>10</sup>B areal density. This makes it easy to reliably establish and verify the presence and microscopic and macroscopic uniformity of the <sup>10</sup>B in the material. Therefore, 90% credit is applied to the minimum <sup>10</sup>B areal density in the criticality calculations, i.e. a 10% penalty is applied. This 10% penalty is considered conservative since there are no significant remaining uncertainties in the <sup>10</sup>B areal density. In Chapter 9 the qualification and on production tests for METAMIC<sup>®</sup> to support 90% <sup>10</sup>B credit are specified. With 90% credit, the target weight percent of boron carbide in METAMIC<sup>®</sup> is 31 for all MPCs, as summarized in Table 1.2.8, consistent with the test coupons used in the EPRI evaluations [1.2.11]. The maximum permitted value is 33.0 wt% to allow for necessary fabrication flexibility.

Because METAMIC<sup>®</sup> is a solid material, there is no capillary path through which spent fuel pool water can penetrate METAMIC<sup>®</sup> panels and chemically react with aluminum in the interior of the material to generate hydrogen. Any chemical reaction of the outer surfaces of the METAMIC<sup>®</sup> neutron absorber panels with water to produce hydrogen occurs rapidly and reduces to an insignificant amount in a short period of time. Nevertheless, combustible gas monitoring for METAMIC<sup>®</sup>-equipped MPCs and purging or exhausting the space under the MPC lid during welding and cutting operations, is required until sufficient field experience is gained that confirms that little or no hydrogen is released by METAMIC<sup>®</sup> during these operations.

Mechanical properties of 31 wt.% METAMIC<sup>®</sup> based on coupon tests of the material in the as-fabricated condition and after 48 hours of an elevated temperature state at 900°F are summarized below from the EPRI report [1.2.11].

<i>Mechanical Properties of 31wt.% B<sub>4</sub>C METAMIC</i>		
<i>Property</i>	<i>As-Fabricated</i>	<i>After 48 hours of 900°F Temperature Soak</i>
<i>Yield Strength (psi)</i>	<i>32937 ± 3132</i>	<i>28744 ± 3246</i>
<i>Ultimate Strength (psi)</i>	<i>40141 ± 1860</i>	<i>34608 ± 1513</i>
<i>Elongation (%)</i>	<i>1.8 ± 0.8</i>	<i>5.7 ± 3.1</i>

*The required flexural strain of the neutron absorber to ensure that it will not fracture when the supporting basket wall flexes due to the worst case lateral inertial loading, has been set at 0.2% for the MPCs. The 1% minimum elongation of 31wt.% B<sub>4</sub>C METAMIC<sup>®</sup> indicated by the above table means that METAMIC<sup>®</sup> will have a minimum factor of safety of five against cracking under the most severe postulated mechanical accident conditions for the MPCs.*

*EPRI's extensive characterization effort [1.2.11], which was focused on 15 and 31 wt.% B<sub>4</sub>C METAMIC<sup>®</sup> served as the principal basis for a recent USNRC SER for 31wt.% B<sub>4</sub>C METAMIC for used in wet storage [1.2.12]. Additional studies on METAMIC<sup>®</sup> [1.2.13], EPRI's and others work provide the confidence that 31wt.% B<sub>4</sub>C METAMIC<sup>®</sup> will perform its intended function in the MPCs.*

#### *1.2.1.3.1.3 Locational Fixity of Neutron Absorbers*

*Both Boral and METAMIC<sup>®</sup> neutron absorber panels are completely enclosed in Alloy X (stainless steel) sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld. Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient evolutions. Finally, the pocket joint detail ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.*

#### 1.2.1.3.2 Neutron Shielding

The specification of the HI-STORM overpack and HI-TRAC transfer cask neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation to appropriate levels;
- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;
- Stability of the homogeneous nature of the shielding material matrix;

- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and
- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

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

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

The HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) will be added to reduce the freezing point for low temperature operations (e.g., below 32°F) [1.2.7].

Neutron shielding in the HI-TRAC 125 and 125D transfer casks in the axial direction is provided by Holtite-A within the top lid. HI-TRAC 125 also contains Holtite-A in the transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal B<sub>4</sub>C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

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

### Density

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

## Hydrogen

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

## Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (nominal) weight concentration. Holtite-A may be specified with a B<sub>4</sub>C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a nominal B<sub>4</sub>C weight percent of 1%.

## Design Temperature

The design temperatures of Holtite-A *are provided in Table 1.B.1. ~~is set at 300°F.~~* The maximum spatial temperatures of Holtite-A under all normal operating conditions must be demonstrated to be below ~~these~~ design temperatures, *as applicable.*

## Thermal Conductivity

The Holtite-A neutron shielding material is stable below the design temperature for the long term and provides excellent shielding properties for neutrons. A conservative, lower bound conductivity is stipulated for use in the thermal analyses of Chapter 4 (Section 4.2) based on information in the technical literature.

### 1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7)<sup>†</sup>. Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the baseplate and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are installed in the ducts (Figures 1.2.8 and 1.2.8A) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets.

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

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<sup>†</sup> The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated. Those overpacks without the shield shell are required to have a higher concrete density in the overpack body to provide compensatory shielding. See Table 1.D.1.

#### 1.2.1.4 Lifting Devices

Lifting of the HI-STORM 100 System may be accomplished either by attachment at the top of the storage overpack ("top lift"), as would typically be done with a crane, or by attachment at the bottom ("bottom lift"), as would be effected by a number of lifting/handling devices.

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The anchor blocks are integrally welded to the overpack radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM100) or inlet air duct horizontal plates (HI-STORM 100S). The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment.

A bottom lift of the HI-STORM 100 storage overpack is effected by the insertion of four hydraulic jacks underneath the inlet vent horizontal plates (Figure 1.2.1). A slot in the overpack baseplate allows the hydraulic jacks to be placed underneath the inlet vent horizontal plate. The hydraulic jacks lift the loaded overpack to provide clearance for inserting or removing a device for transportation.

The standard design HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket trunnions. The HI-TRAC 125D is equipped with only lifting trunnions. The lifting trunnions are positioned just below the top forging. The two pocket trunnions are located above the bottom forging and attached to the outer shell. The pocket trunnions are designed to allow rotation of the HI-TRAC. All trunnions are built from a high strength alloy with proven corrosion and non-galling characteristics. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The lifting trunnions are installed by threading into tapped holes just below the top forging.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered through the HI-TRAC transfer cask using lifting cleats. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

#### 1.2.1.5 Design Life

The design life of the HI-STORM 100 System is 40 years. This is accomplished by using material of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 9, is also implemented to ensure the HI-STORM 100 System will exceed its design life of 40 years. The design considerations that assure the HI-STORM 100 System performs as designed throughout the service life include the following:

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

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

## MPC

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

The adequacy of the HI-STORM 100 System for its design life is discussed in Sections 3.4.11 and 3.4.12.

### 1.2.2 Operational Characteristics

#### 1.2.2.1 Design Features

The HI-STORM 100 System incorporates some unique design improvements. These design innovations have been developed to facilitate the safe long term storage of SNF. Some of the design originality is discussed in Subsection 1.2.1 and below.

The free volume of the MPCs is inerted with 99.995% pure helium gas during the spent nuclear fuel loading operations. Table 1.2.2 specifies the helium fill requirements for the MPC internal cavity.

The HI-STORM overpack has been designed to synergistically combine the benefits of steel and concrete. The steel-concrete-steel construction of the HI-STORM overpack provides ease of fabrication, increased strength, and an optimal radiation shielding arrangement. The concrete is primarily provided for radiation shielding and the steel is primarily provided for structural functions.

The strength of concrete in tension and shear is conservatively neglected. Only the compressive strength of the concrete is accounted for in the analyses.

The criticality control features of the HI-STORM 100 are designed to maintain the neutron multiplication factor  $k$ -effective (including uncertainties and calculational bias) at less than 0.95 under all normal, off-normal, and accident conditions of storage as analyzed in Chapter 6. This level of conservatism and safety margins is maintained, while providing the highest storage capacity.



### 1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STORM 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the ~~Design~~ Drawings in Section 1.5. A summary of the general actions needed for the loading and unloading operations is provided below. Figures 1.2.16 and 1.2.17 provide a pictorial view of typical loading and unloading operations, respectively.

#### Loading Operations

At the start of loading operations, the HI-TRAC transfer cask is configured with the pool lid installed. The HI-TRAC water jacket is filled with demineralized water or a 25% ethylene glycol solution depending on the ambient temperature conditions. The lift yoke is used to position HI-TRAC in the designated preparation area or setdown area for HI-TRAC inspection and MPC insertion. The annulus is filled with plant demineralized water (~~borated if necessary~~), and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with water (*borated if necessary*). Based on the MPC model and fuel enrichment, (~~as required by the CoC~~), this may be borated water or plant demineralized water (*see Section 2.1*). HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

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

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. ~~Dose rates are measured at the MPC lid and around the mid-height circumference of HI-TRAC to ensure that the dose rates are within expected values.~~ The Automated Welding System baseplate shield (if used) is installed to reduce dose rates around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS) or other approved welding process. Liquid penetrant examinations are performed on the root and final passes. A multi-layer liquid penetrant or volumetric examination is also performed on the MPC lid-to-shell weld. ~~The water level is raised~~

to the top of the MPC and the weld is hydrostatically tested. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid to shell) to verify weld integrity and to ensure that leakage rates are within acceptance criteria. The MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line. *At the appropriate time in the sequence of activities, based on the type of test performed (hydrostatic or pneumatic), a pressure test of the MPC enclosure vessel is performed.*

For storage of moderate burnup fuel MPCs containing all moderate burnup fuel, a Vacuum Drying System (VDS) may be used to remove moisture from the MPC cavity. The VDS is connected to the MPC and is used to remove liquid water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated. This process is continued until the pressure in the MPC meets the technical specification limit and can be held there for the required amount of time.

For storage of high burnup fuel, and as an option for storage of moderate burnup fuel, the reduction of residual moisture in the MPC to trace amounts is accomplished using a Forced Helium Dehydration (FHD) system, as described in Appendix 2.B. Relatively warm and dry helium is recirculated through the MPC cavity, which helps maintain the SNF in a cooled condition while moisture is being removed. The warm, dry gas is supplied to the MPC drain port and circulated through the MPC cavity where it absorbs moisture. The humidified gas travels out of the MPC and through appropriate equipment to cool and remove the absorbed water from the gas. The dry gas may be heated prior to its return to the MPC in a closed loop system to accelerate the rate of moisture removal in the MPC. This process is continued until the temperature of the gas exiting the demoinsturizing module described in Appendix 2.B meets the *specified* limit. ~~specified in the technical specifications.~~

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

The MPC closure ring is then placed on the MPC, aligned, tacked in place, and seal welded, providing redundant closure of the MPC lid and cover plates confinement closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in the annulus is drained. The AWS Baseplate shield is removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and HI-

TRAC dose rates are measured. The HI-TRAC top lid is installed and the bolts are torqued. The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point of the MPC.

Rigging is installed between the MPC lift cleats and the lift yoke. The rigging supports the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. For the standard design transfer cask, the HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and rigging support the MPC during the transfer operations.

MPC transfer from the HI-TRAC transfer cask into the overpack may be performed inside or outside the fuel building. Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways. The loaded HI-TRAC may be handled in the vertical or horizontal orientation. The loaded HI-STORM can only be handled vertically.

For MPC transfers inside the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100 overpack, the vent duct shield inserts installed. If using HI-TRAC 125D, the HI-STORM mating device is secured to the top of the empty overpack (Figure 1.2.18). The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM, or the mating device, as applicable. After the HI-TRAC is positioned atop the HI-STORM or secured to the mating device, as applicable, the MPC is raised slightly. With the standard HI-TRAC design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and the HI-TRAC is prepared for removal from on top of HI-STORM (with HI-TRAC 125D, the transfer cask must first be disconnected from the mating device). For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and/or mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs are installed and torqued.

For MPC transfers outside of the fuel building, the empty HI-STORM overpack is inspected and staged with the lid removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts installed. For HI-TRAC 125D, the mating device is secured to the top of the overpack. The loaded HI-TRAC is transported to the cask transfer facility in the vertical or horizontal orientation. A number of methods may be utilized as long as the handling limitations prescribed in the technical specifications are not exceeded.

To place the loaded HI-TRAC in a horizontal orientation, a transport frame or “cradle” is utilized. If the cradle is equipped with rotation trunnions they are used to engage the HI-TRAC

100 or 125 pocket trunnions. While the loaded HI-TRAC is lifted by the lifting trunnions, the HI-TRAC is lowered onto the cradle rotation trunnions. Then, the crane lowers and the HI-TRAC pivots around the pocket trunnions and is placed in the horizontal position in the cradle.

The HI-TRAC 125D does not include pocket trunnions in its design. Therefore, the user must downend the transfer cask onto the transport frame using appropriately designed rigging in accordance with the site's heavy load control program.

If the loaded HI-TRAC is transferred to the cask transfer facility in the horizontal orientation, the HI-TRAC transport frame and/or cradle are placed on a transport vehicle. The transport vehicle may be an air pad, railcar, heavy-haul trailer, dolly, etc. If the loaded HI-TRAC is transferred to the cask transfer facility in the vertical orientation, the HI-TRAC may be lifted by the lifting trunnions or seated on the transport vehicle. During the transport of the loaded HI-TRAC, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms.

*For MPCs containing any HBF, the Supplemental Cooling System (SCS) is required to be operational during the time the loaded and backfilled MPC is in HI-TRAC to ensure fuel cladding temperatures remain within limits. The SCS is discussed in detail in Section 4.5 and the design criteria for the system are provided in Appendix 2.C. The SCS is not required when the MPC is inside the overpack, regardless of decay heat load.*

After the loaded HI-TRAC arrives at the cask transfer facility, the HI-TRAC is upended by a crane if the HI-TRAC is in a horizontal orientation. The loaded HI-TRAC is then placed, using the crane located in the transfer area, on top of HI-STORM, which has been inspected and staged with the lid removed, vent duct shield inserts installed, the alignment device positioned, and the mating device installed, as applicable.

After the HI-TRAC is positioned atop the HI-STORM or the mating device, the MPC is raised slightly. In the standard design, the transfer lid door locking pins are removed and the doors are opened. With the HI-TRAC 125D, the pool lid is removed using the mating device. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. For the HI-STORM 100, the doors are closed and HI-TRAC is removed from on top of HI-STORM or disconnected from the mating device, as applicable. For the HI-STORM 100S, the standard design HI-TRAC may need to be lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location. The MPC lift cleats and slings are removed from atop the MPC. The alignment device, vent duct shield inserts, and mating device is/are removed, as applicable. The pool lid is removed from the mating device and re-attached to the HI-TRAC 125D prior to its next use. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs and nuts are installed.

After the HI-STORM has been loaded either within the fuel building or at a dedicated cask

transfer facility, the HI-STORM is then moved to its designated position on the ISFSI pad. The HI-STORM overpack may be moved using a number of methods as long as the handling limitations listed in the technical specifications are not exceeded. The loaded HI-STORM must be handled in the vertical orientation, and may be lifted from the top by the anchor blocks or from the bottom by the inlet vents. After the loaded HI-STORM is lifted, it may be placed on a transport mechanism or continue to be lifted by the lid studs and transported to the storage location. The transport mechanism may be an air pad, crawler, railcar, heavy-haul trailer, dolly, etc. During the transport of the loaded HI-STORM, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms. Once in position at the storage pad, vent operability testing is performed to ensure that the system is functioning within its design parameters.

In the case of HI-STORM 100A, the anchor studs are installed and fastened into the anchor receptacles in the ISFSI pad in accordance with the design requirements.

### Unloading Operations

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

The MPC is recovered from HI-STORM either at the cask transfer facility or the fuel building using any of the methodologies described in Section 8.1. The HI-STORM lid is removed, the alignment device positioned, and, for the HI-STORM 100, the vent duct shield inserts are installed, and the MPC lift cleats are attached to the MPC. For HI-TRAC 125D, the mating device is installed. Rigging is attached to the MPC lift cleats. For the HI-STORM 100S and the standard HI-TRAC design, the transfer doors may need to be opened to avoid interfering with the MPC lift cleats. For HI-TRAC 125D, the mating device (possibly containing the pool lid) is secured to the top of the overpack. HI-TRAC is raised and positioned on top of HI-STORM or secured to the mating device, as applicable. For HI-TRAC 125D, the pool lid is ensured to be out of the transfer path for the MPC. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the standard design HI-TRAC transfer lid doors are closed and the locking pins are installed. For HI-TRAC 125D, the pool lid is installed and the transfer cask is unsecured from the mating device. HI-TRAC is removed from on top of HI-STORM. *As required based on the presence of high burnup fuel, the Supplemental Cooling System is installed and placed into operation.*

The HI-TRAC is brought into the fuel building and, for the standard design, manipulated for bottom lid replacement. The transfer lid is replaced with the pool lid. The MPC lift cleats and rigging support the MPC during lid transfer operations.

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the rigging, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water (borated, if necessary). The annulus and HI-TRAC top surfaces are protected from debris that will be produced when removing the MPC lid.

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established around the MPC ports. The RVOAs are attached to the vent and drain port. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is cooled using ~~a closed-loop heat exchanger~~ *to appropriate means, if necessary*, to reduce the MPC internal temperature to allow water flooding. Following the fuel cool-down, the MPC is flooded with borated or unborated water, *as required*. ~~in accordance with the CoC~~. The MPC lid-to-MPC shell weld is removed. Then, all weld removal equipment is removed with the MPC lid left in place.

The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris. HI-TRAC and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and HI-TRAC are decontaminated in preparation for re-utilization.

### 1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

#### 1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorbing materials in the fuel basket. The MPC-24/24E/24EF, ~~MPC-24E, and 24EF~~ (all with lower enriched fuel) and the MPC-68/68F/68FF do not rely on soluble boron credit during loading or the assurance that water cannot enter the MPC during storage to meet the stipulated criticality limits.

Each MPC model is equipped with ~~Boral~~ neutron absorber plates affixed to the fuel cell walls as shown on the ~~design~~ drawings in Section 1.5. The minimum  $^{10}\text{B}$  areal density specified for the ~~Boral~~ neutron absorber in each MPC model is shown in Table 1.2.2. These values are chosen to be consistent with the assumptions made in the criticality analyses.

The MPC-24, MPC-24E and 24EF (all with higher enriched fuel) and the MPC-32 and MPC-32F take credit for soluble boron in the MPC water for criticality prevention during wet loading and unloading operations. Boron credit is only necessary for these PWR MPCs during loading and unloading operations that take place under water. During storage, with the MPC cavity dry and sealed from the environment, criticality control measures beyond the fixed neutron poisons affixed to the storage cell walls are not necessary because of the low reactivity of the fuel in the dry, helium filled canister and the design features that prevent water from intruding into the canister during storage.

#### 1.2.2.3.2 Chemical Safety

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

#### 1.2.2.3.3 Operation Shutdown Modes

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

#### 1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is seal welded, *non-destructively examined* and *leak pressure* tested. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM overpack exit vents in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 ~~and the Technical Specifications in Appendix A to the CoC~~ for additional details.

#### 1.2.2.3.5 Maintenance Technique

Because of their passive nature, the HI-STORM 100 System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 9 describes the acceptance criteria and maintenance program set forth for the HI-STORM 100.

### 1.2.3 Cask Contents

The HI-STORM 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key *system data and design* parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. ~~and the Approved Contents section of Appendix B to the CoC.~~ This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in ~~the CoC~~ *Table 1.0.1*. A summary of the types of fuel authorized for storage in each MPC model is provided below. All fuel assemblies, *non-fuel hardware*, and *neutron sources* must meet the fuel specifications provided in ~~Appendix B to the CoC~~ *Section 2.1*. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers.

#### MPC-24

The MPC-24 is designed to accommodate up to twenty-four (24) PWR fuel assemblies classified

as intact fuel assemblies, with or without non-fuel hardware.

#### MPC-24E

The MPC-24E is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

#### MPC-24EF

The MPC-24EF is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4).

#### MPC-32

The MPC-32 is designed to accommodate up to thirty-two (32) PWR fuel assemblies ~~classified as intact fuel assemblies~~, with or without non-fuel hardware. *Up to eight (8) of these assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).*

#### MPC-32F

*The MPC-32F is designed to store up to thirty two (32) PWR fuel assemblies with or without non-fuel hardware. Up to eight (8) of these assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32 (see Figure 1.2.3).*

#### MPC-68

The MPC-68 is designed to accommodate up to sixty-eight (68) BWR intact and/or damaged fuel assemblies, with or without channels. For the Dresden Unit 1 or Humboldt Bay plants, the number of damaged fuel assemblies may be up to a total of 68. For damaged fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the number of damaged fuel assemblies is limited to sixteen (16) and must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2).

#### MPC-68F

The MPC-68F is designed to accommodate up to sixty-eight (68) Dresden Unit 1 or Humboldt



Bay BWR fuel assemblies (with or without channels) made up of any combination of fuel assemblies classified as intact fuel assemblies, damaged fuel assemblies, and up to four (4) fuel assemblies classified as fuel debris.

#### MPC-68FF

The MPC-68FF is designed to accommodate up to sixty-eight (68) BWR fuel assemblies with or without channels. Any number of these fuel assemblies may be Dresden Unit 1 or Humboldt Bay BWR fuel assemblies classified as intact fuel or damaged fuel. Dresden Unit 1 and Humboldt Bay fuel debris is limited to eight (8) DFCs. DFCs containing Dresden Unit 1 or Humboldt Bay fuel debris may be stored in any fuel storage location. For BWR fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the total number of fuel assemblies classified as damaged fuel assemblies or fuel debris is limited to sixteen (16), with up to eight (8) of the 16 fuel assemblies classified as fuel debris. These fuel assemblies must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2). The balance of the fuel storage locations may be filled with intact BWR fuel assemblies, up to a total of 68.

Table 1.2.1

KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	7 8	4 5 for PWR 3 for BWR
MPC storage capacity <sup>†</sup> :	MPC-24  MPC-24E  MPC-24EF	Up to 24 intact <del>Zircaloy-ZR</del> or stainless steel clad PWR fuel assemblies with or without non-fuel hardware. Up to four damaged fuel assemblies may be stored in the MPC-24E and up to four (4) damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24EF
	MPC-32  MPC-32F	OR  Up to 32 intact <del>Zircaloy-ZR</del> or stainless steel clad PWR fuel assemblies <i>with or without non-fuel hardware. Up to 8 damaged fuel assemblies may be stored in the MPC-32 and up to 8 damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-32F.</i>
	MPC-68	Any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68. For damaged fuel other than Dresden Unit 1 and Humboldt Bay, the number of fuel assemblies is limited to 16, with the balance being intact fuel assemblies.  OR

<sup>†</sup> See Section 2.11.2.3 and Appendix B to the CoC for a complete description of cask *authorized cask* contents and fuel specifications, respectively.

Table 1.2.1 (continued)  
KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
MPC storage capacity:	MPC-68F	Up to 4 damaged fuel containers with <del>Zircaloy</del> -ZR clad Dresden Unit 1 (D-1) or Humboldt Bay (HB) BWR fuel debris and the complement damaged <del>Zircaloy</del> -ZR clad Dresden Unit 1 or Humboldt Bay BWR fuel assemblies in damaged fuel containers or intact Dresden Unit 1 or Humboldt Bay BWR intact fuel assemblies.
	MPC-68FF	OR Up to 68 Dresden Unit 1 or Humboldt Bay intact fuel or damaged fuel and up to 8 damaged fuel containers containing D-1 or HB fuel debris. For other BWR plants, up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris with the complement intact fuel assemblies, up to a total of 68. The number of damaged fuel containers containing BWR fuel debris is limited to eight (8) for all BWR plants.

Table 1.2.2  
KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725 <sup>o†</sup> /-40 <sup>o††</sup>	725 <sup>o†</sup> /-40 <sup>o††</sup>
Design internal pressure (psig)		
Normal conditions	100	100
Off-normal conditions	<del>100</del> 110	<del>100</del> 110
Accident Conditions	200	200
Total heat load, max. (kW)	28.74	28.19
Maximum permissible peak fuel cladding temperature:		
Long Term Normal (°F)	See Table 2.2.3752	See Table 2.2.3752
Short Term Operations (°F)	752 or 1058 <sup>†††</sup>	752 or 1058 <sup>†††</sup>
Off-normal and Accident (°F)	1058	1058
MPC internal environment helium fill (99.995% fill helium purity)	(all pressure ranges are at a reference temperature of 70°F)	(all pressure ranges are at a reference temperature of 70°F)
MPC-24 (heat load ≤ 27.77 kW)	≥ 29.3 psig and ≤ 33.3 psig OR 0.1212 +/-10% g-moles/liter	
MPC-24E/24EF (heat load ≤ 28.17 kW)	≥ 29.3 psig and ≤ 33.3 psig OR 0.1212 +/-10% g-moles/liter	
MPC-68/68F/68FF (heat load ≤ 28.19 kW)		0.1218 +/-10% g-moles/liter OR ≥ 29.3 psig and ≤ 33.3 psig
MPC-32/32F (heat load ≤ 28.74 kW)	≥ 29.3 psig and ≤ 33.3 psig OR 0.1212 +/-10% g-moles/liter	
Maximum permissible multiplication factor (k <sub>eff</sub> ) including all uncertainties and biases	< 0.95	< 0.95

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

†† Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2.2 and no fuel decay heat load.

††† See Section 4.5 for discussion of the applicability of the 1058°F temperature limit during MPC drying.

*Table 1.2.2 (cont'd)*  
**KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS**

	PWR	BWR
<i>Fixed Neutron Absorber Boral</i> <sup>10</sup> B Areal Density (g/cm <sup>2</sup> )	0.0267/0.0223 (MPC-24)	0.0372/0.0310 (MPC-68 & MPC-68FF)
<i>Boral/METAMIC</i>	0.0372/0.0310 (MPC-24E, MPC-24EF & MPC-32, & MPC-32F)	0.01/NA (MPC-68F) ( <i>See Note 1</i> )
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

*NOTES:*

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

Table 1.2.3

INTENTIONALLY DELETED

Table 1.2.4

INTENTIONALLY DELETED

Table 1.2.5

INTENTIONALLY DELETED



Table 1.2.6

## HI-STORM 100 OPERATIONS SEQUENCE

Site-specific handling and operations procedures will be prepared, reviewed, and approved by each owner/user.	
1	HI-TRAC and MPC lowered into the fuel pool without lids
2	Fuel assemblies transferred into the MPC fuel basket
3	MPC lid lowered onto the MPC
4	HI-TRAC/MPC assembly moved to the decon pit and MPC lid welded in place, volumetrically or multi-layer PT examined, <del>hydrostatically tested</del> , and <del>leak</del> <i>pressure</i> tested
5	MPC dewatered, moisture removed, backfilled with helium, and the closure ring welded
6	HI-TRAC annulus drained and external surfaces decontaminated
7	MPC lifting cleats installed and MPC weight supported by rigging
8	HI-TRAC pool lid removed and transfer lid attached (not applicable to HI-TRAC 125D)
9	MPC lowered and seated on HI-TRAC transfer lid (not applicable to HI-TRAC 125D)
9a	HI-STORM mating device secured to top of empty HI-STORM overpack (HI-TRAC 125D only)
10	HI-TRAC/MPC assembly transferred to atop HI-STORM overpack or mating device, as applicable
11	MPC weight supported by rigging and transfer lid doors opened (standard design HI-TRAC) or pool lid removed (HI-TRAC 125D)
12	MPC lowered into HI-STORM overpack, and HI-TRAC removed from atop HI-STORM overpack/mating device
12a	HI-STORM mating device removed (HI-TRAC 125D only)
13	HI-STORM overpack lid installed and bolted in place
14	HI-STORM overpack placed in storage at the ISFSI pad
15	For HI-STORM 100A (or 100SA) users, the overpack is anchored to the ISFSI pad by installation of nuts onto studs and torquing to the minimum required torque.

Table 1.2.7

REPRESENTATIVE ASME BOLTING AND THREADED ROD MATERIALS ACCEPTABLE  
FOR THE HI-STORM 100A ANCHORAGE SYSTEM

ASME MATERIALS FOR BOLTING

Composition	I.D.	Type Grade or UNC No.	Ultimate Strength (ksi)	Yield Strength (ksi)	Code Permitted Size Range <sup>†</sup>
C	SA-354	BC K04100	125	109	$t \leq 2.5''$
$\frac{3}{4}$ Cr	SA-574	51B37M	170	135	$t \geq 5/8''$
1 Cr – 1/5 Mo	SA-574	4142	170	135	$t \geq 5/8''$
1 Cr-1/2 Mo-V	SA-540	B21 (K 14073)	165	150	$t \leq 4''$
5 Cr – 1/2 Mo	SA-193	B7	125	105	$t \leq 2.5''$
2Ni – 3/4 Cr – 1/4 Mo	SA-540	B23 (H-43400)	135	120	
2Ni – 3/4 Cr – 1/3 Mo	SA-540	B-24 (K-24064)	135	120	
17Cr-4Ni-4Cu	SA-564	630(H-1100)	140	115	
17Cr-4Ni-4Cu	SA-564	630(H-1075)	145	125	
25Ni-15Cr-2Ti	SA-638	660	130	85	
22ER-13Ni-5Mn	SA-479	XM-19(S20910)	135	105	

Note: The materials listed in this table are representative of acceptable materials and have been abstracted from the ASME Code, Section II, Part D, Table 3. Other materials listed in the Code are also acceptable as long as they meet the size requirements, the minimum requirements on yield and ultimate strength (see Table 2.0.4), and are suitable for the environment.

<sup>†</sup> Nominal diameter of the bolt (or rod) as listed in the Code tables. Two-inch diameter studs/rods are specified for the HI-STORM 100A.

Table 1.2.8

*METAMIC<sup>®</sup> DATA FOR HOLTEC MPCs*

<i>MPC Type</i>	<i>Min. B-10 areal density required by criticality analysis (g/cm<sup>2</sup>)</i>	<i>Nominal Weight Percent of B<sub>4</sub>C and Reference METAMIC<sup>®</sup> Panel Thickness</i>			
		<i>100% Credit</i>	<i>90% Credit</i>	<i>75% Credit</i>	<i>Ref. Thickness (inch)</i>
<i>MPC-24</i>	<i>0.020</i>	<i>27.6</i>	<i>31</i>	<i>37.2</i>	<i>0.075</i>
<i>MPC-68, -68FF, -32, -32F, -24E, and -24EF</i>	<i>0.0279</i>	<i>27.8</i>	<i>31</i>	<i>37.4</i>	<i>0.104</i>

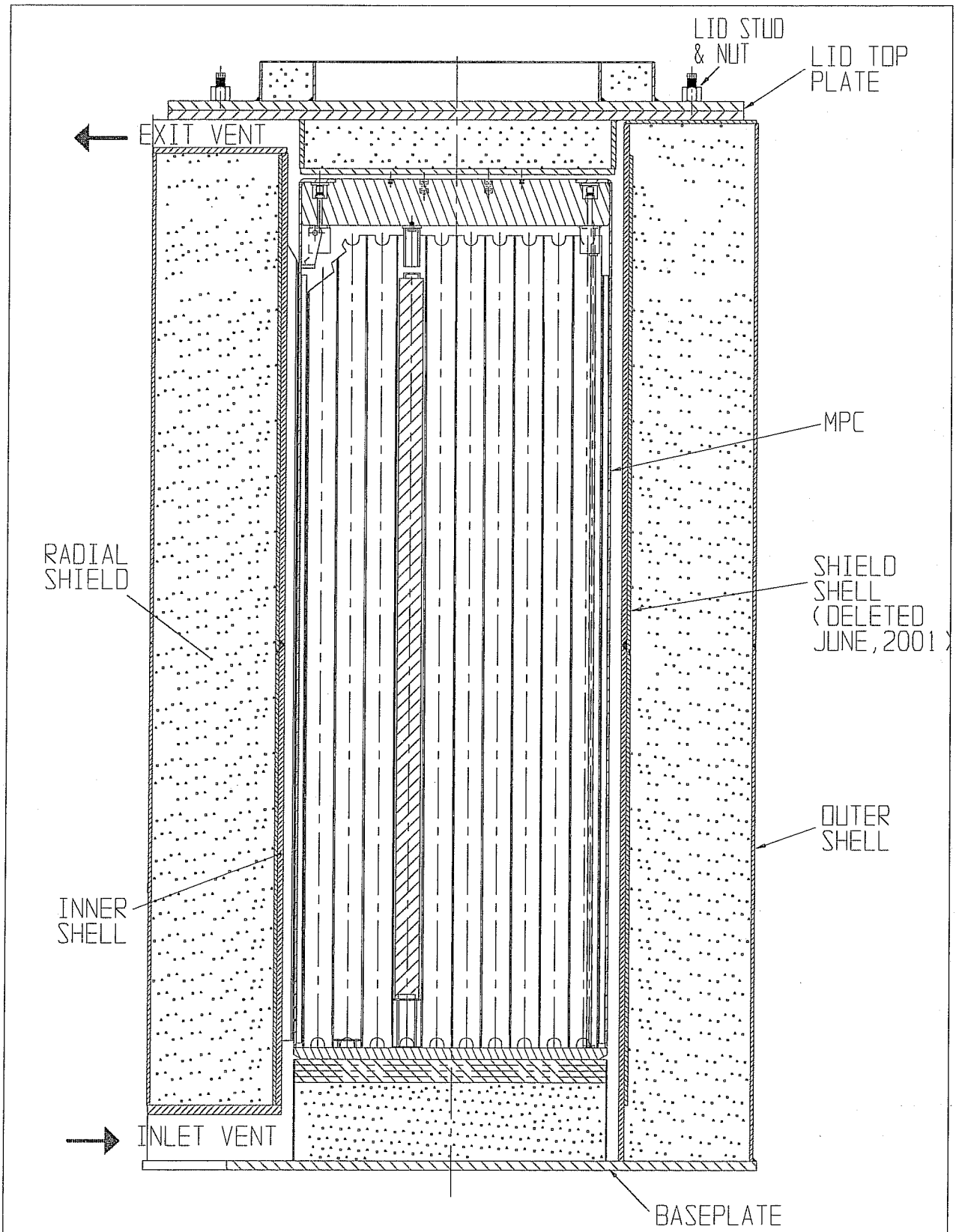


FIGURE 1.2.1; CROSS SECTION VIEW OF THE HI-STORM 100 SYSTEM

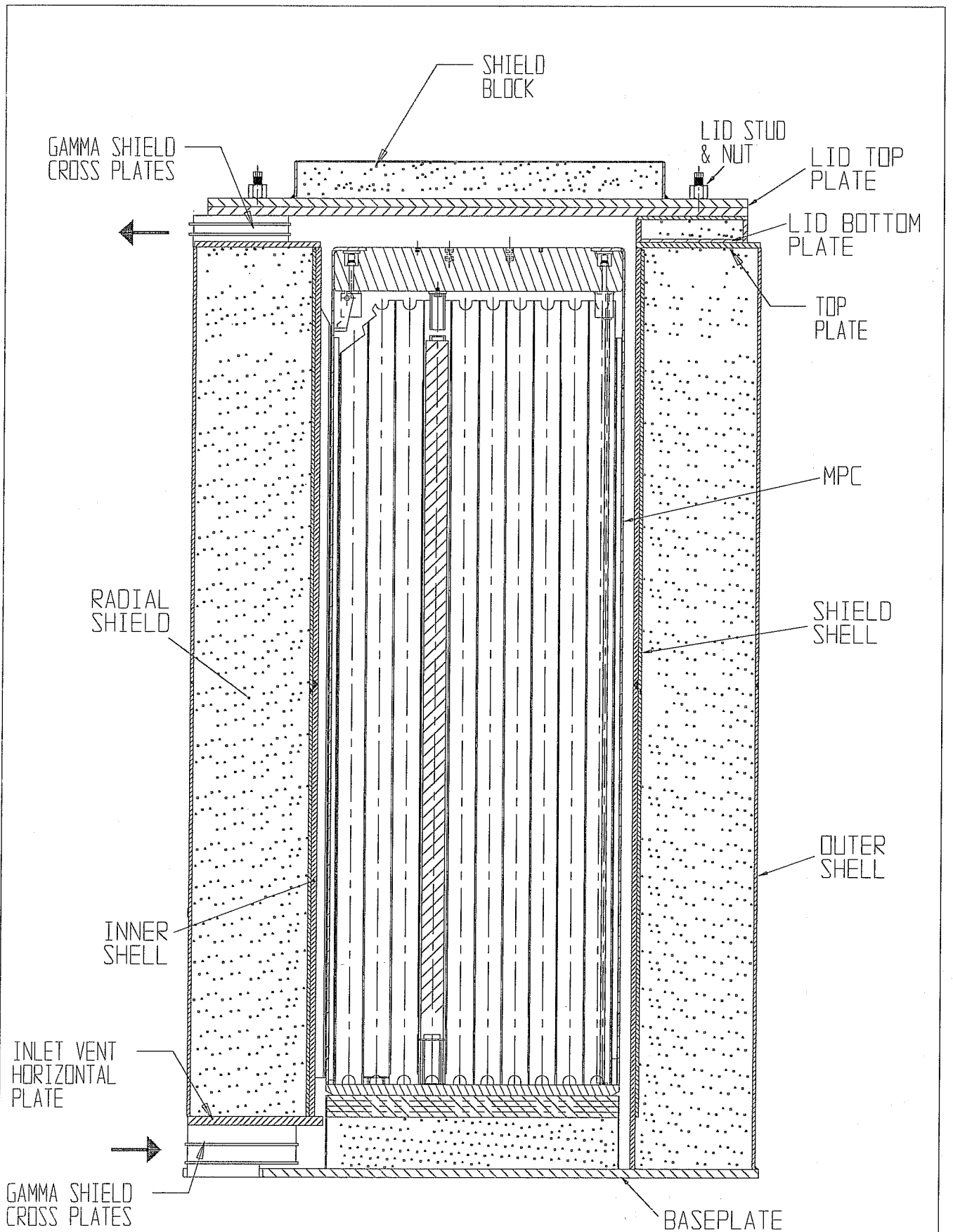


FIGURE 1.2.1A; CROSS SECTION VIEW OF THE HI-STORM 100S SYSTEM

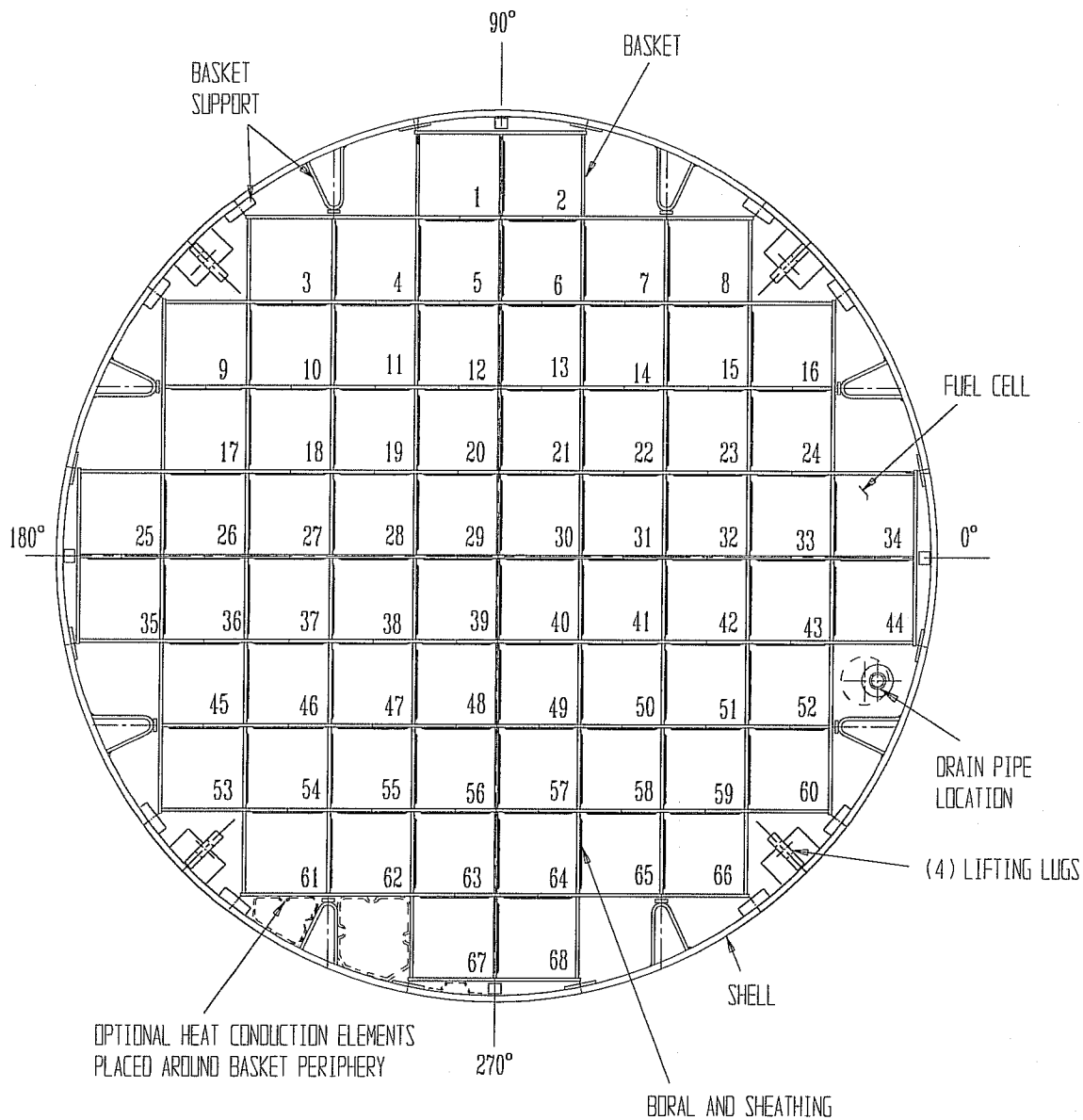


FIGURE 1.2.2; MPC-68 CROSS SECTION VIEW

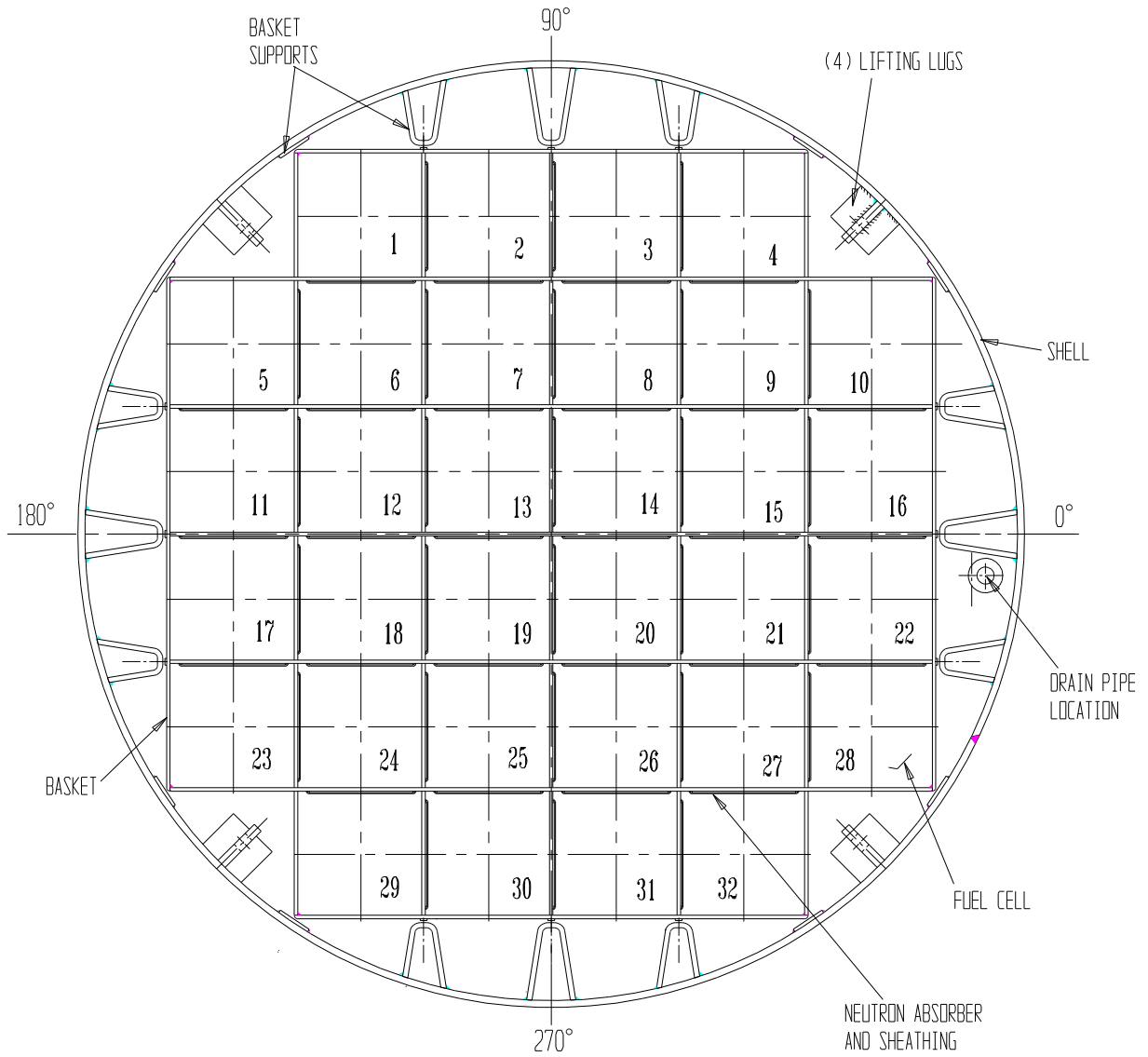


FIGURE 1.2.3; MPC-32/32F CROSS SECTION

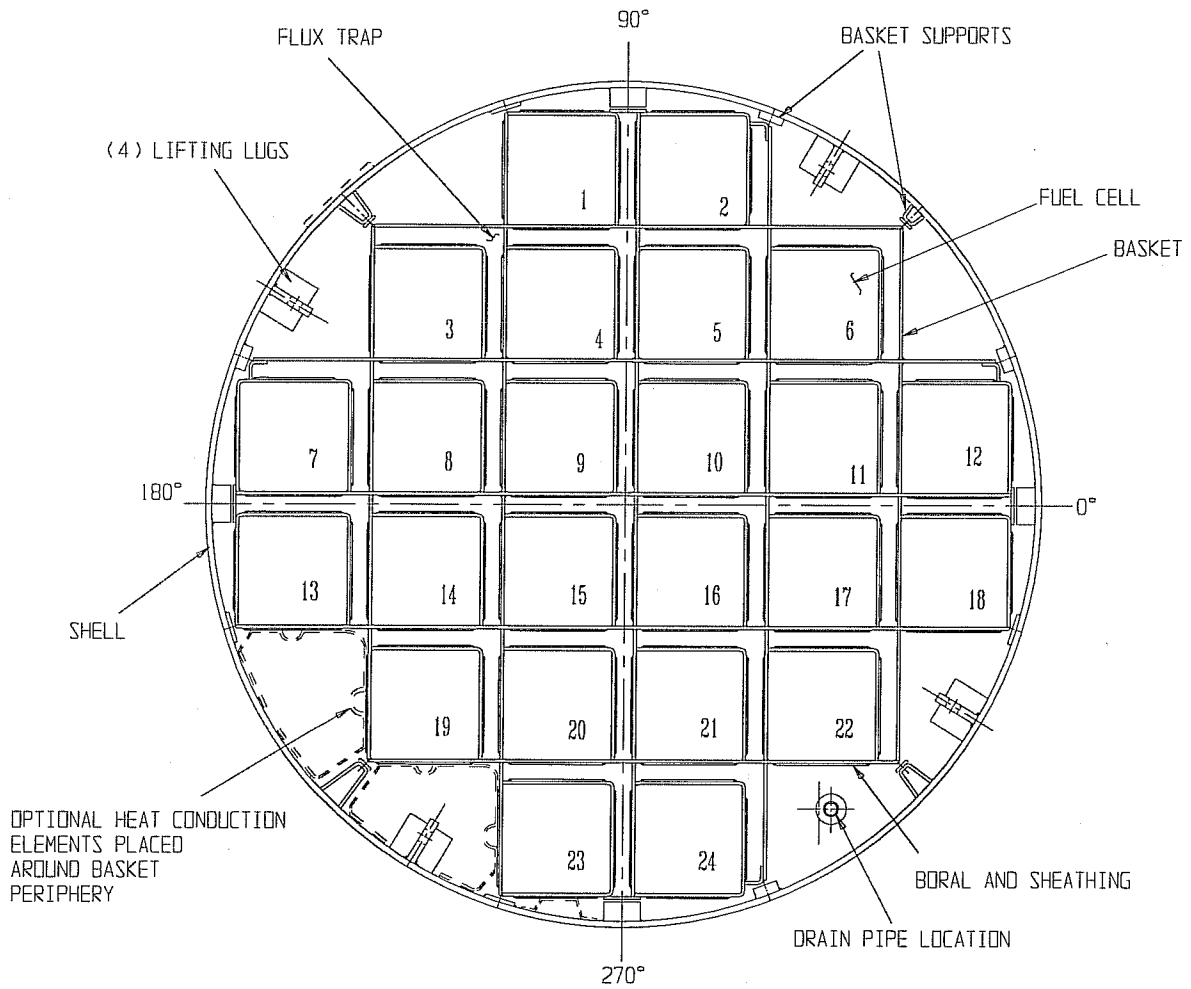


FIGURE 1.2.4; MPC-24/24E/24EF CROSS SECTION VIEW



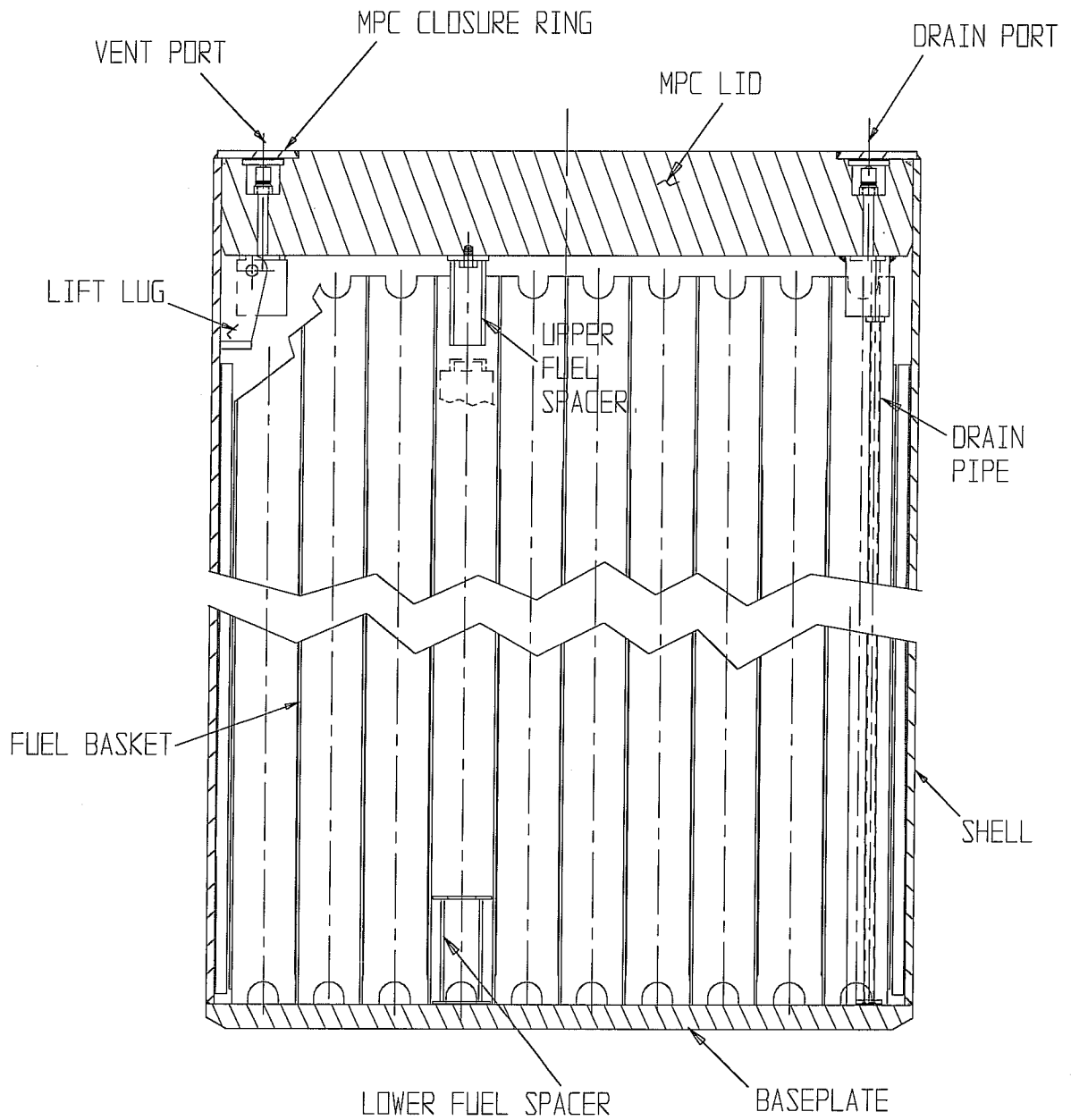


FIGURE 1.2.5; CROSS SECTION ELEVATION VIEW OF MPC

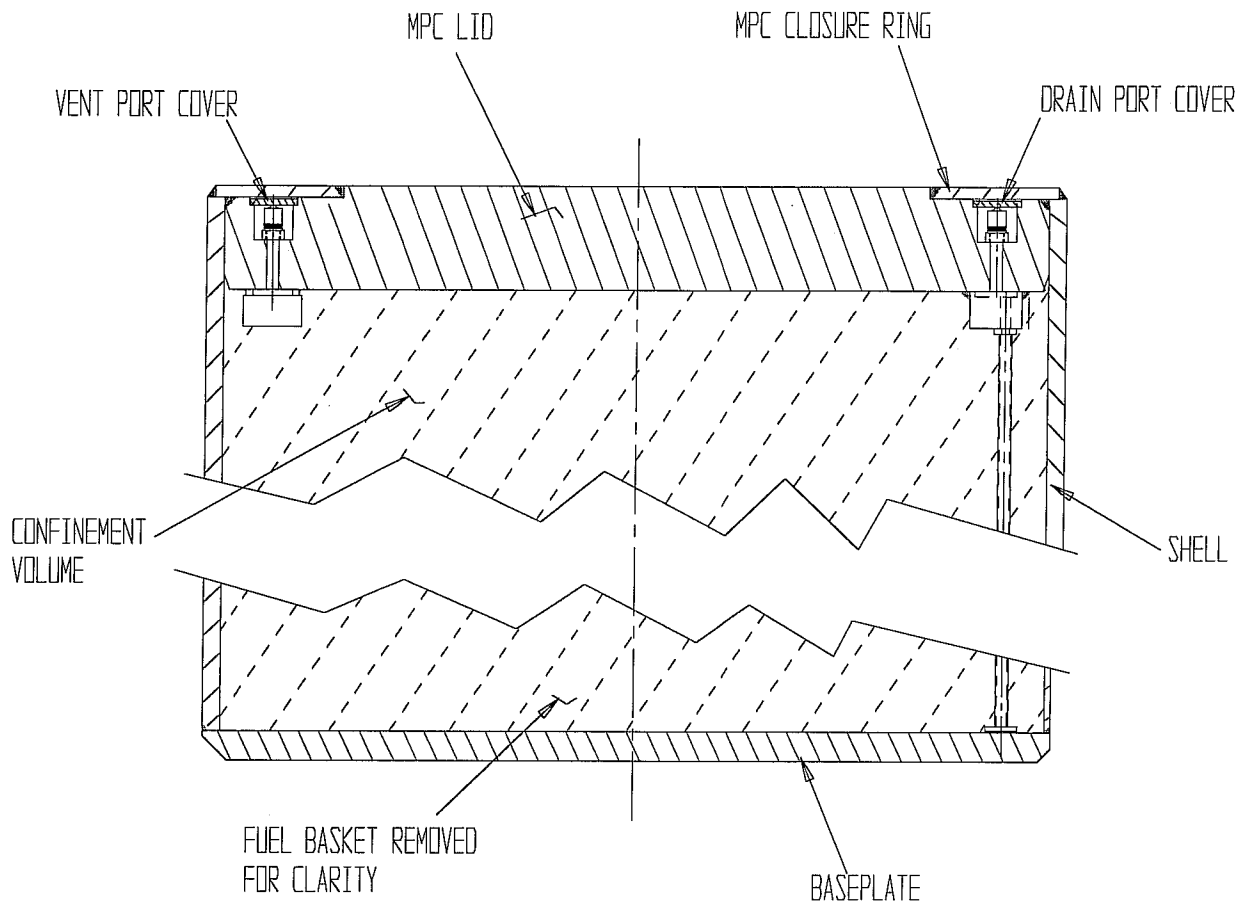


FIGURE 1.2.6; MPC CONFINEMENT BOUNDARY

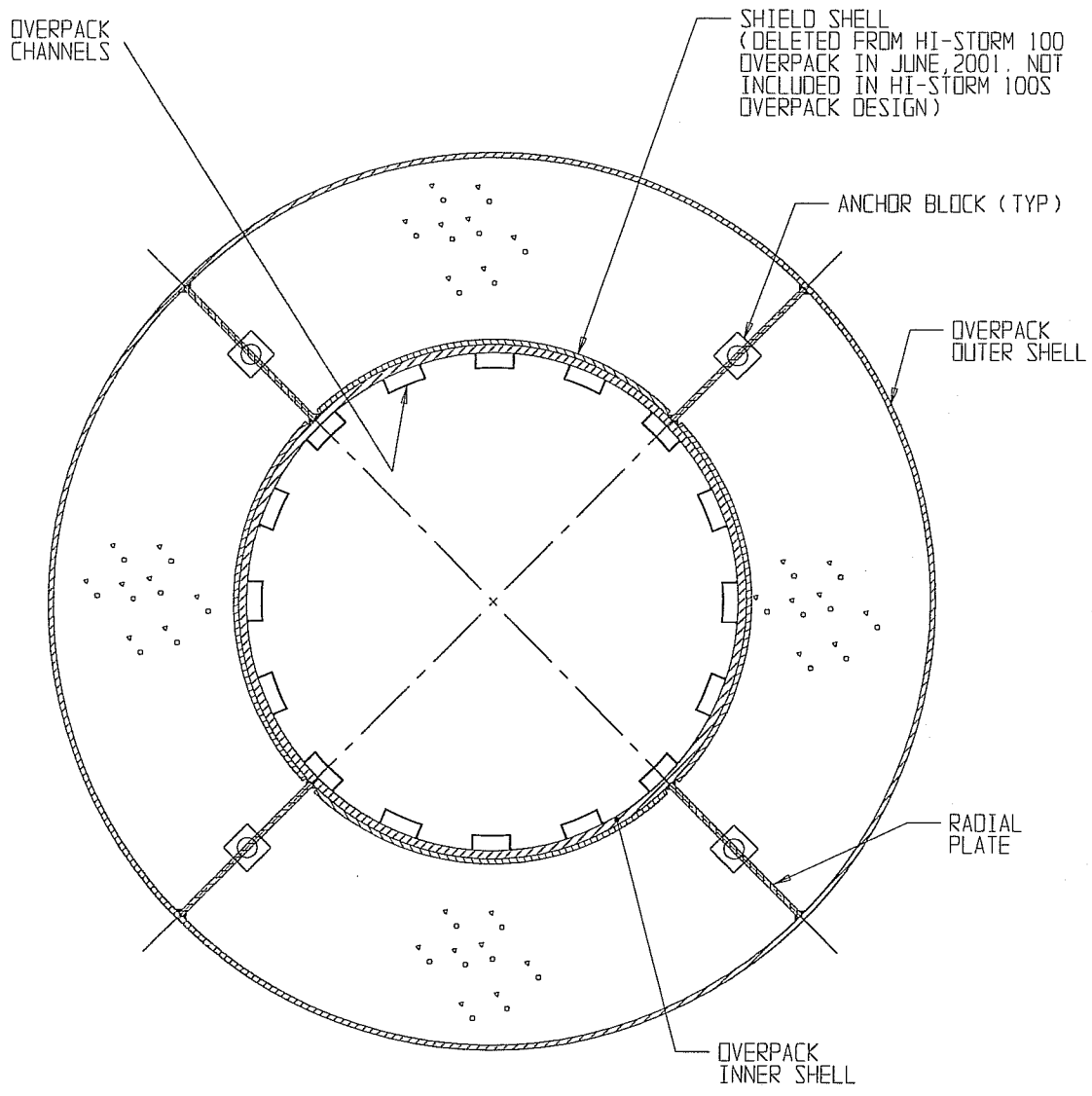


FIGURE 1.2.7; CROSS SECTION OF HI-STORM OVERPACK

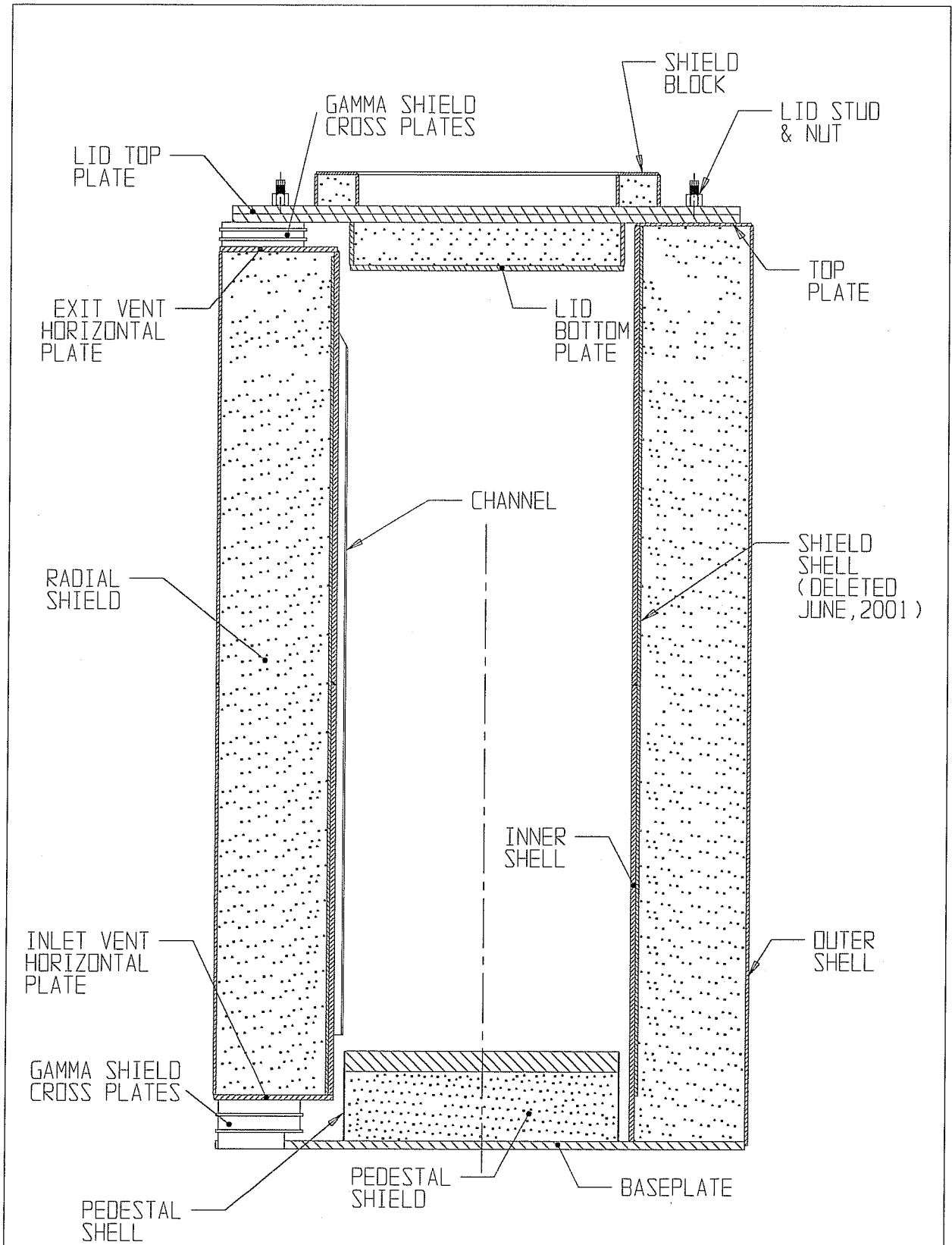


FIGURE 1.2.8; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

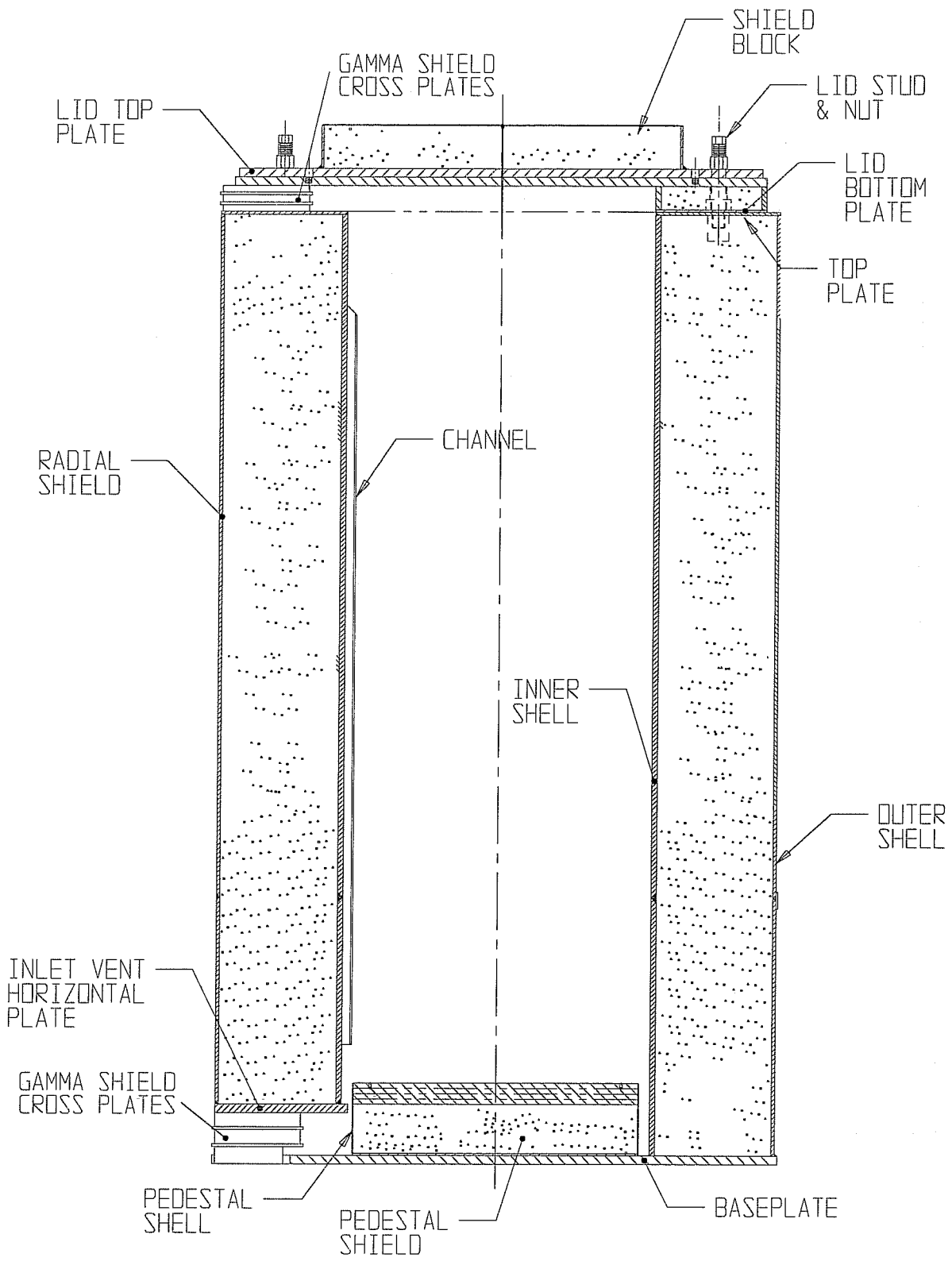


FIGURE 1.2.8A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

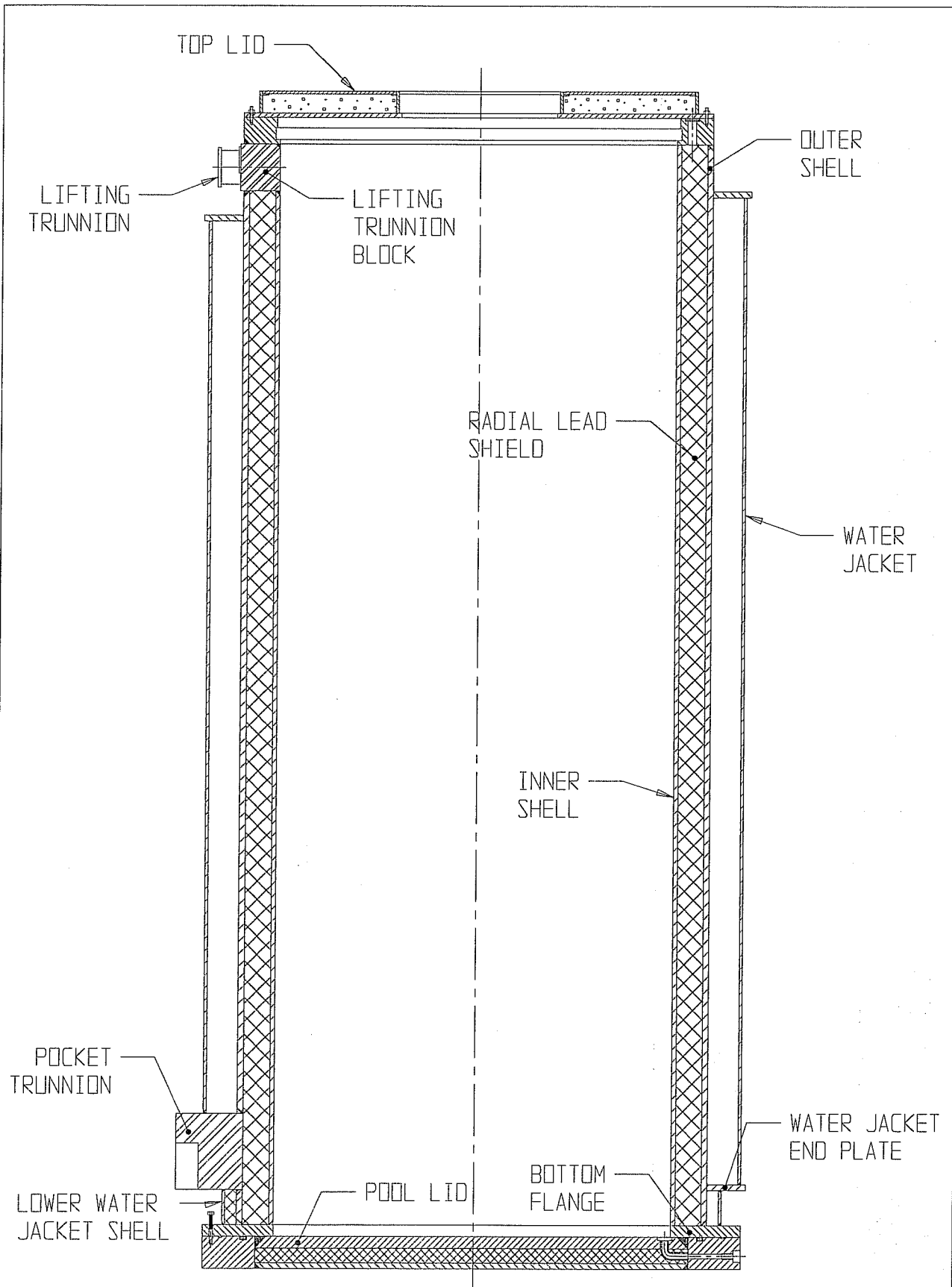


FIGURE 1.2.9; 125TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW

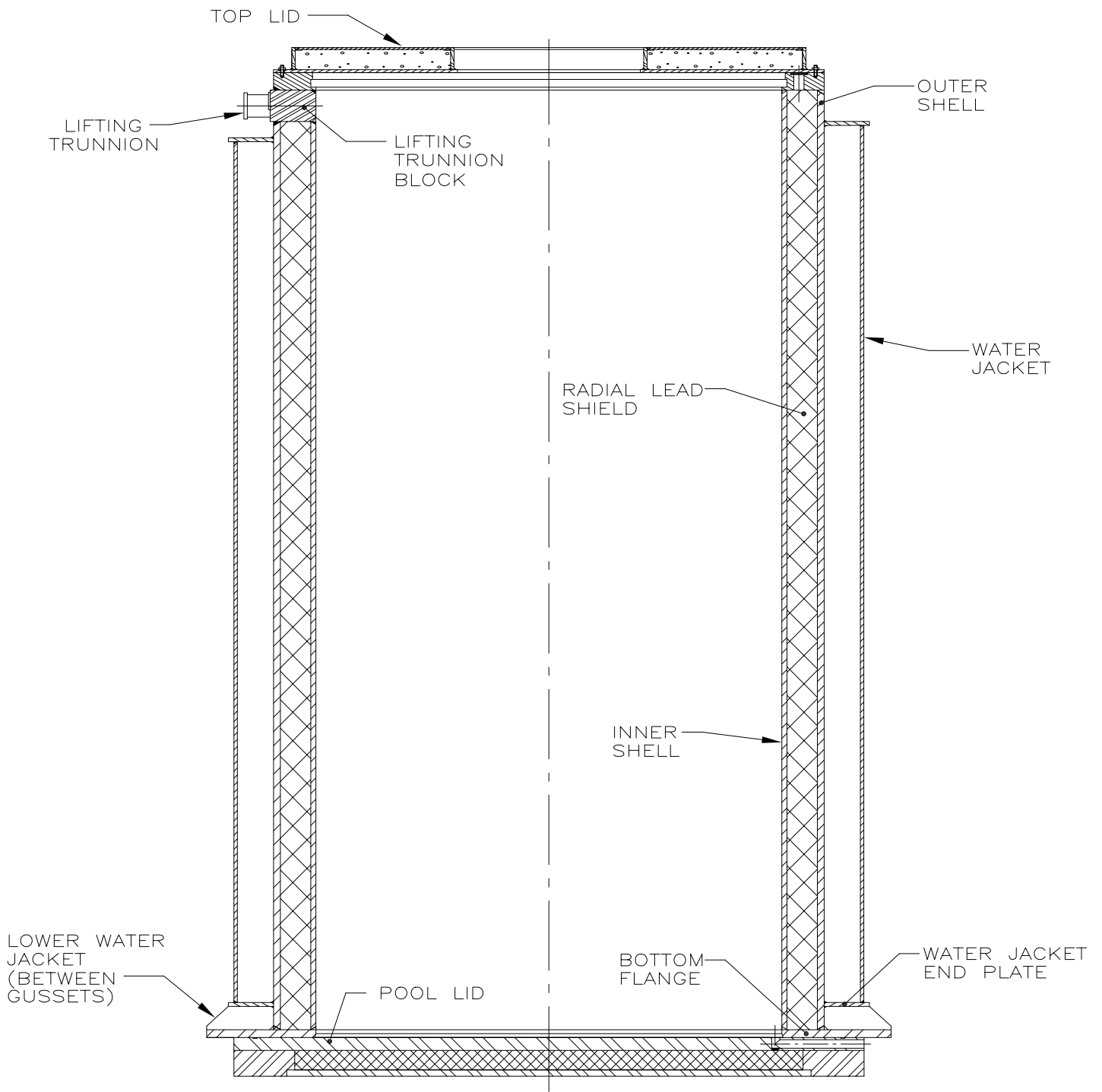


FIGURE 1.2.9A; HI-TRAC 125D TRANSFER CASK  
CROSS SECTIONAL ELEVATION VIEW

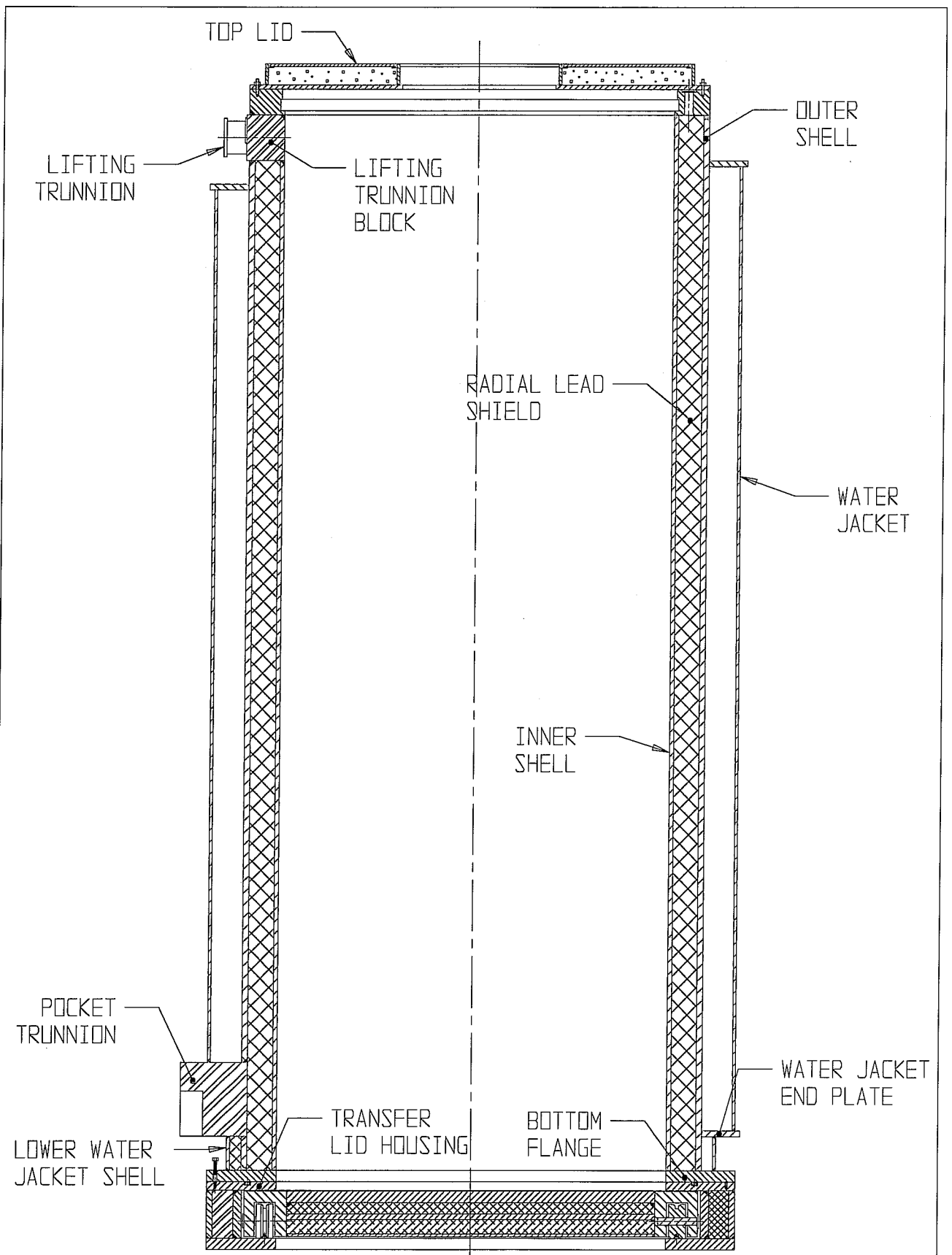


FIGURE 1.2.10; 125TON HI-TRAC TRANSFER CASK WITH TRANSFER LID  
 CROSS SECTIONAL ELEVATION VIEW



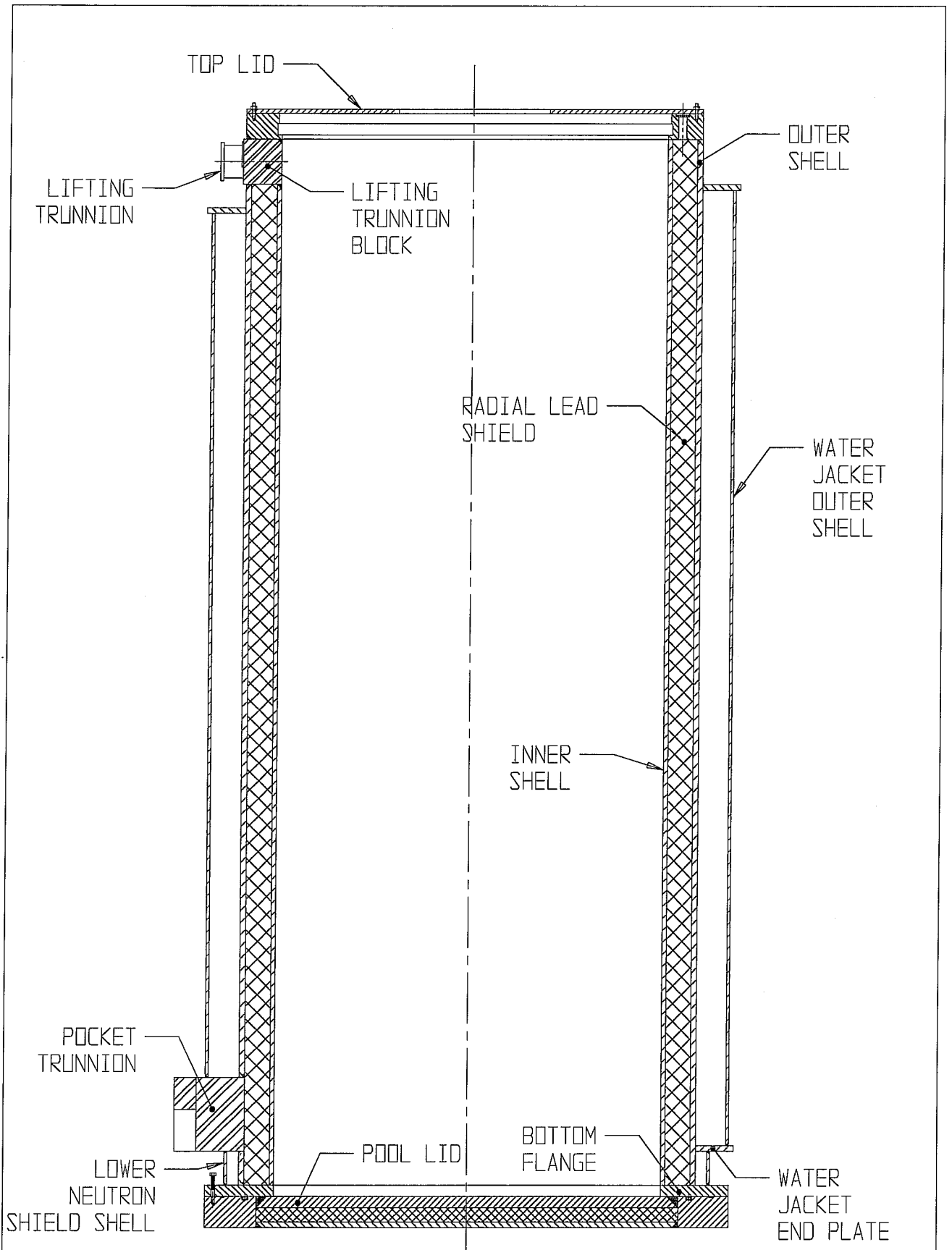


FIGURE 1.2.11; 100 HI-TRAC TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW

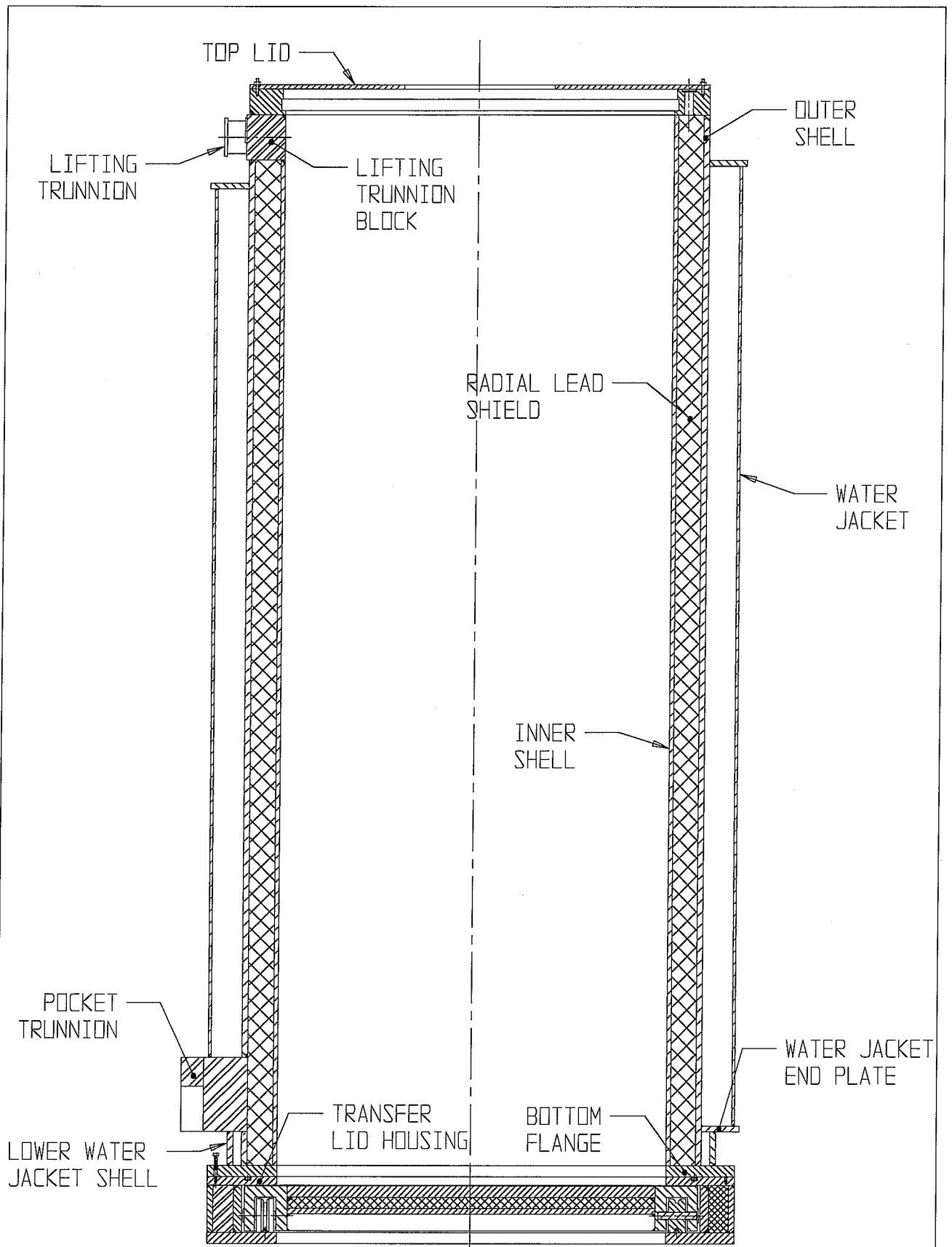


FIGURE 1.2.12; 100 TON HI-TRAC TRANSFER CASK WITH TRANSFER LID CROSS SECTIONAL ELEVATION VIEW

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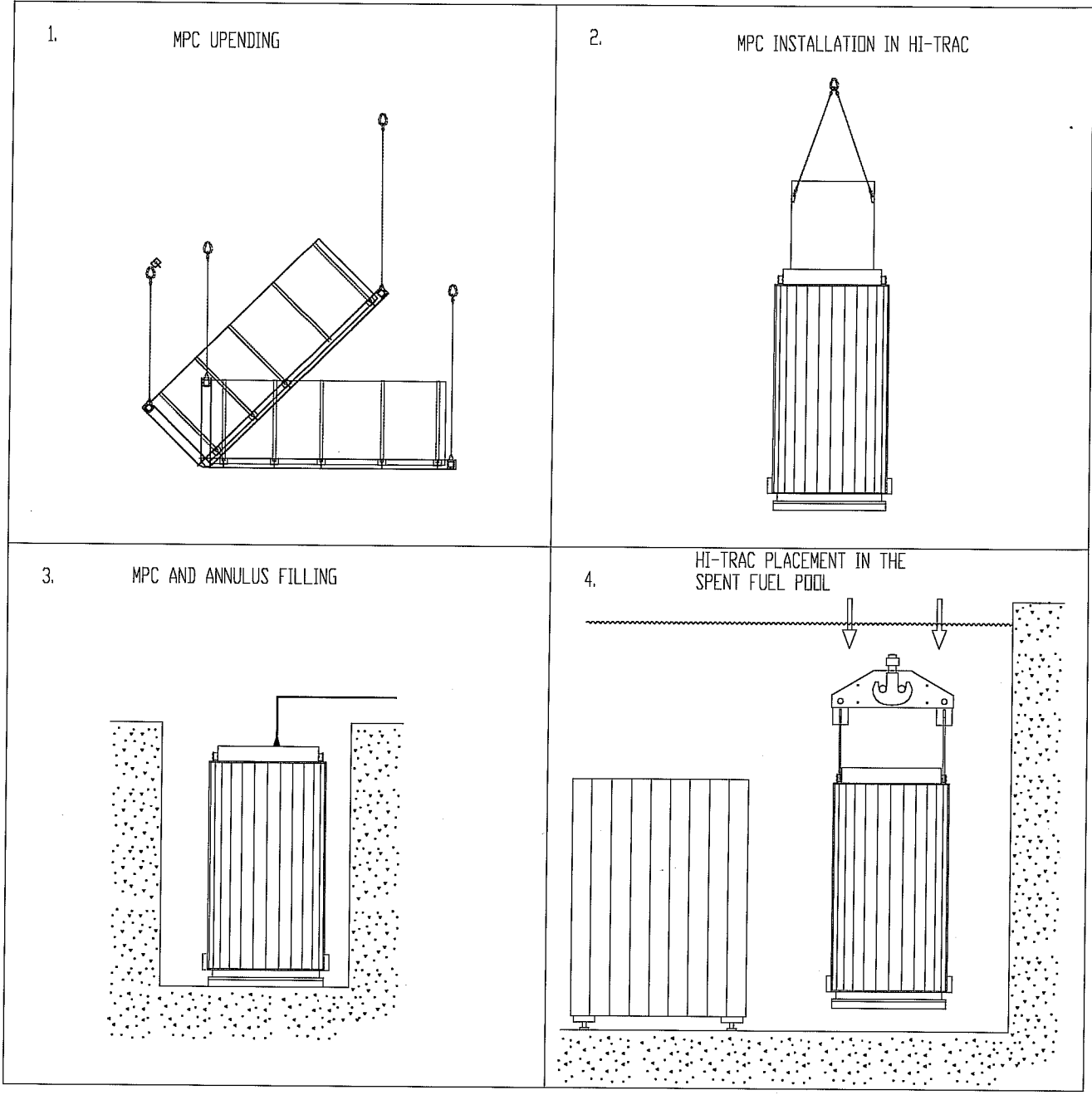
FIGURE 1.2.13; DELETED

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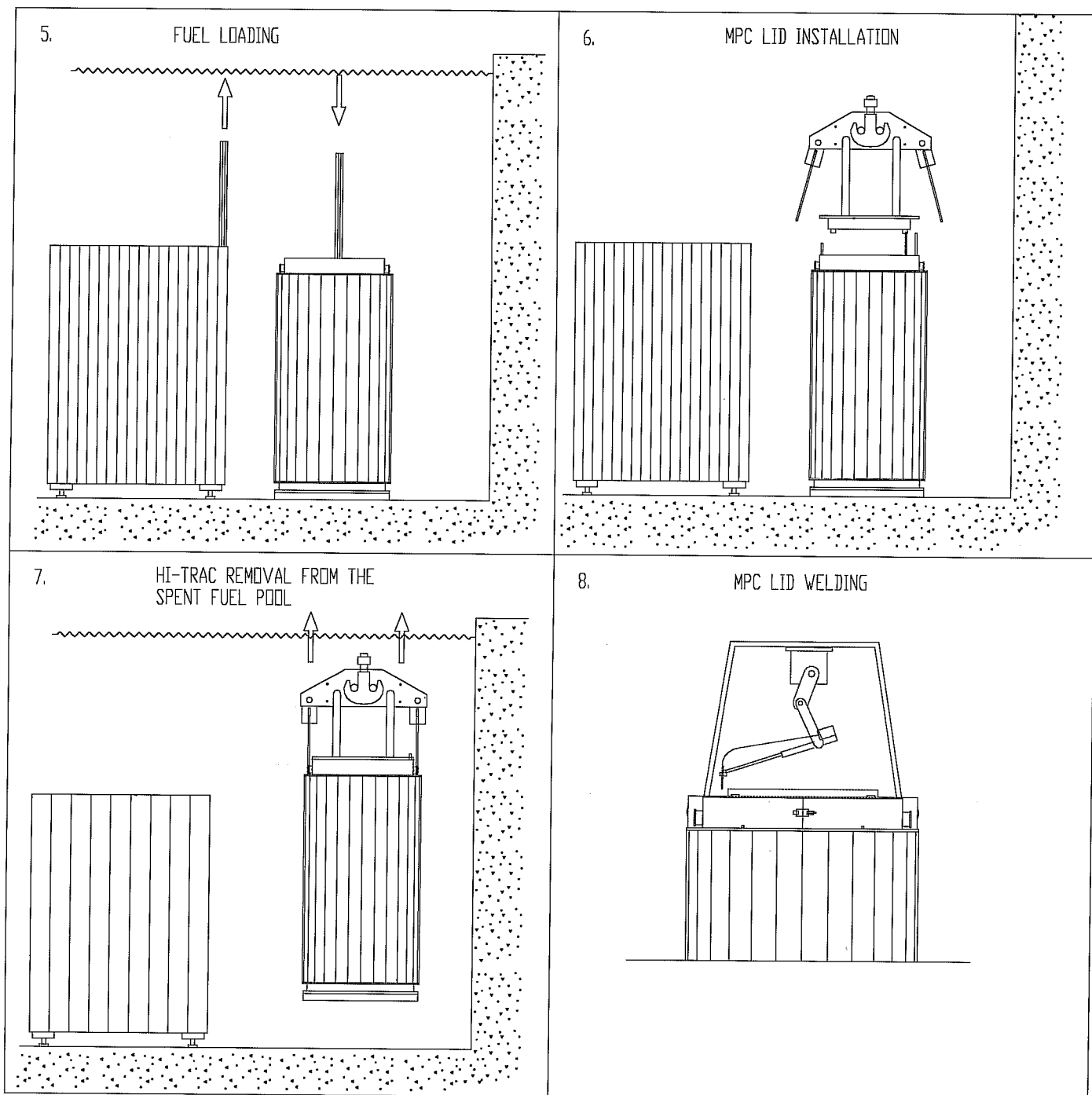
FIGURE 1.2.14; DELETED

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FIGURE 1.2.15; DELETED



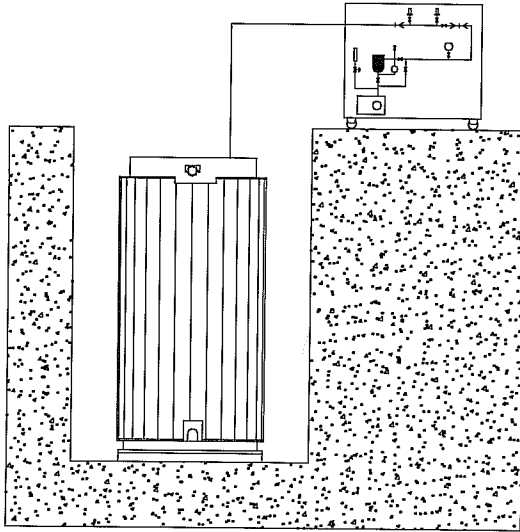
**Figure 1.2.16a; Major HI-STORM 100 Loading Operations (Sheet 1 of 6)**



**Figure 1.2.16b; Major HI-STORM 100 Loading Operations (Sheet 2 of 6)**

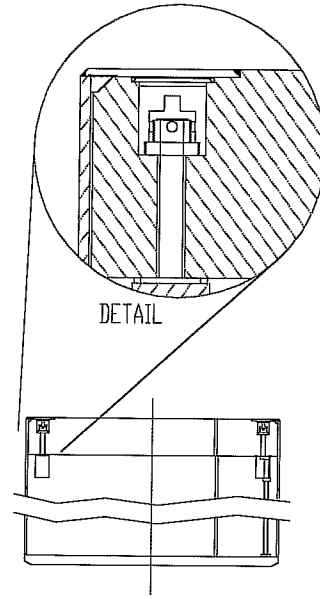
9.

MPC DRAINING/DRYING AND BACKFILLING



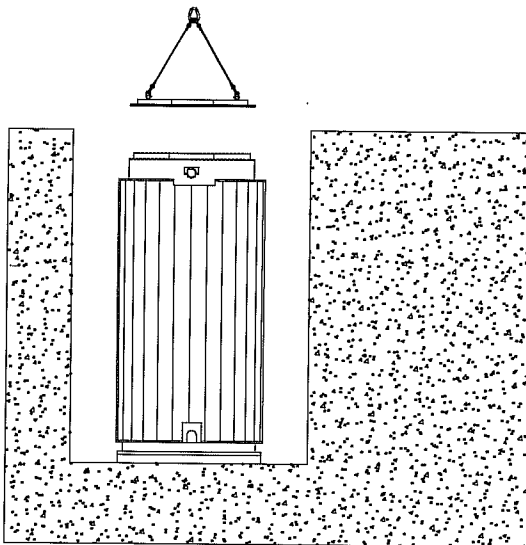
10.

MPC VENT/DRAIN PORT AND CLOSURE RING DETAILS



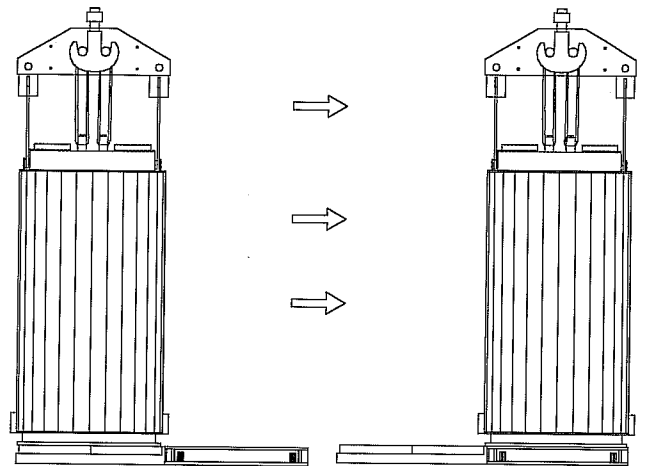
11.

HI-TRAC TOP LID INSTALLATION



12.

BOTTOM LID REPLACEMENT

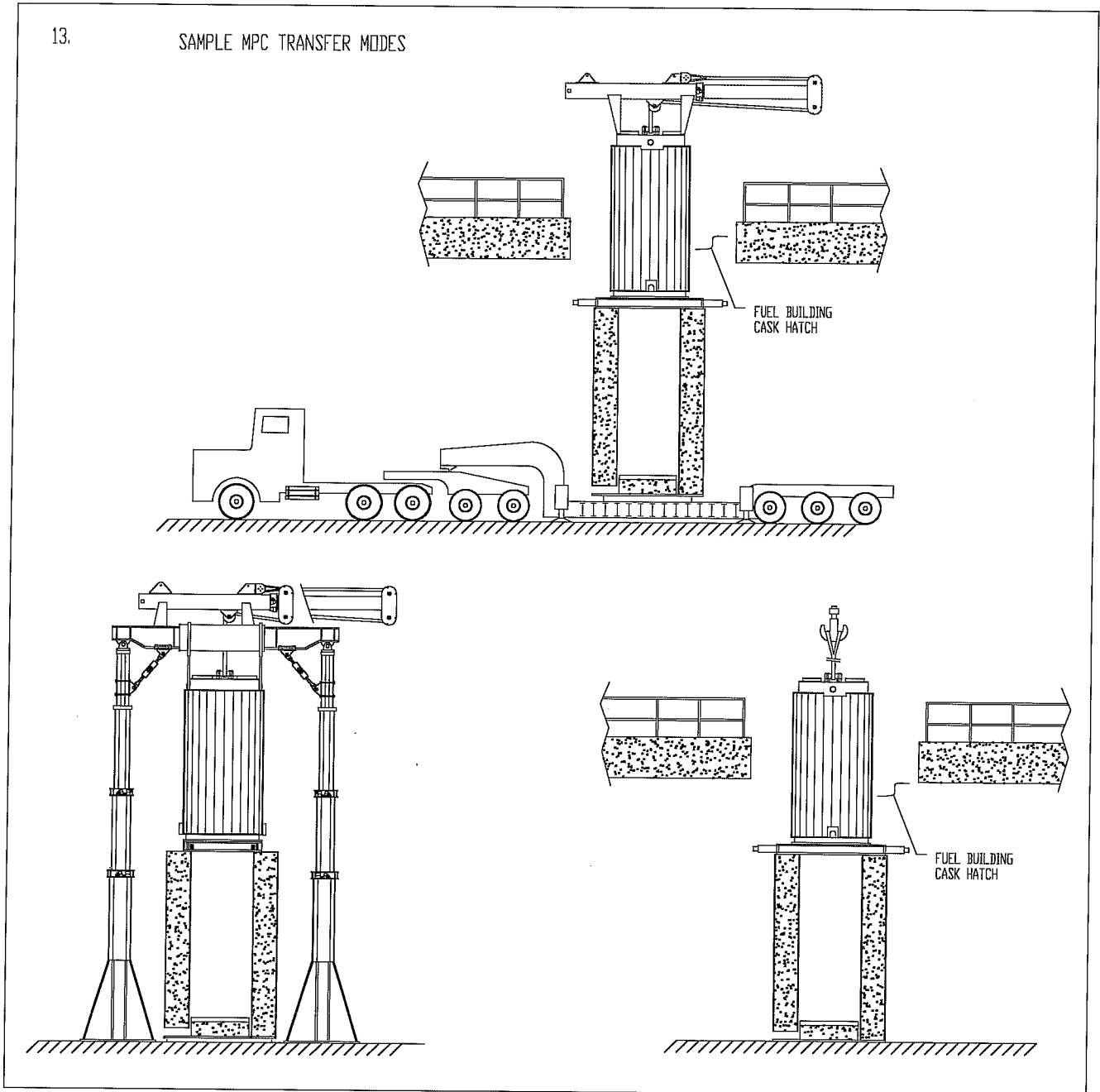


**Figure 1.2.16c; Major HI-STORM 100 Loading Operations (Sheet 3 of 6)**



13.

SAMPLE MPC TRANSFER MODES

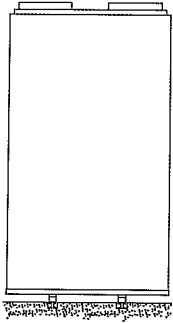


**Figure 1.2.16d; Major HI-STORM 100 Loading Operations (Sheet 4 of 6)**

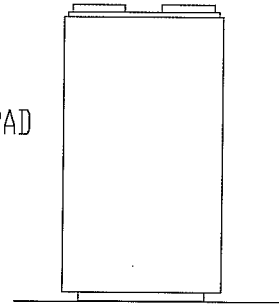
14.

SAMPLE HI-STORM HANDLING METHODS

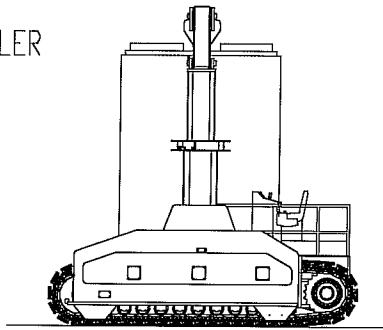
RAIL DOLLY



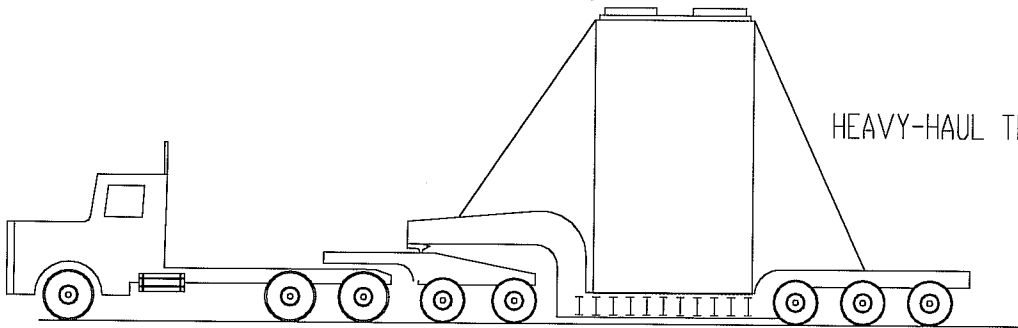
AIR PAD



CASK CRAWLER



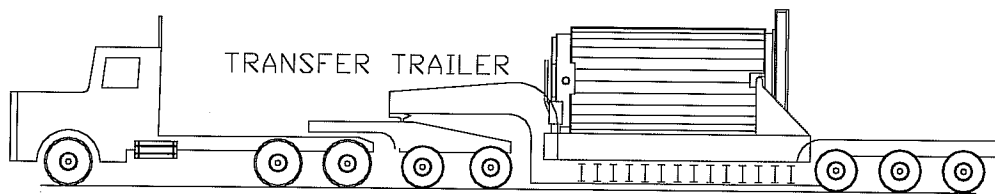
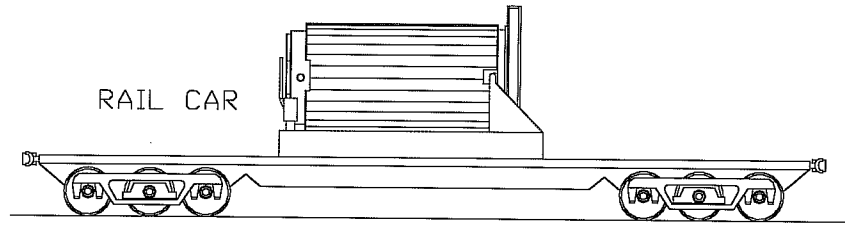
HEAVY-HAUL TRAILER



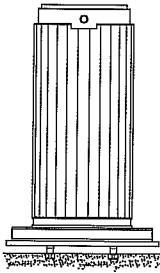
**Figure 1.2.16e; Example of HI-STORM 100 Handling Options (Sheet 5 of 6)**

15.

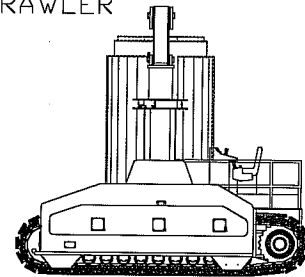
SAMPLE HI-TRAC HANDLING METHODS



RAIL DOLLY

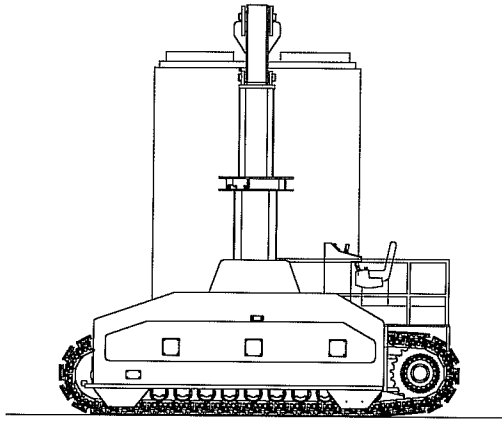


VERTICAL CASK CRAWLER

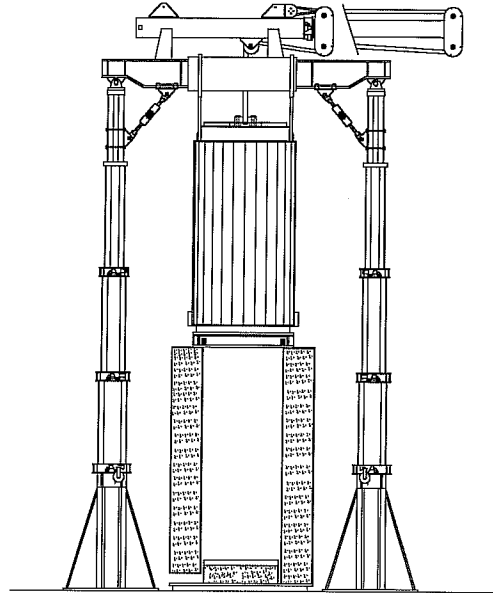


**Figure 1.2.16f; Example of HI-TRAC Handling Options (Sheet 6 of 6)**

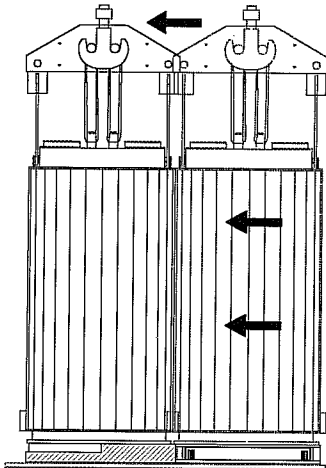
1. HI-STORM TRANSFER TO THE OVERHEAD LIFTING DEVICE (CRAWLER SHOWN)



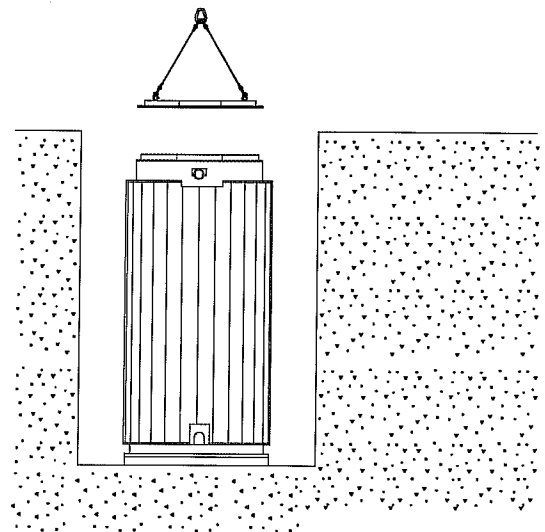
2. MPC TRANSFER INTO HI-TRAC (HYDRAULIC LIFTING GANTRY SHOWN)



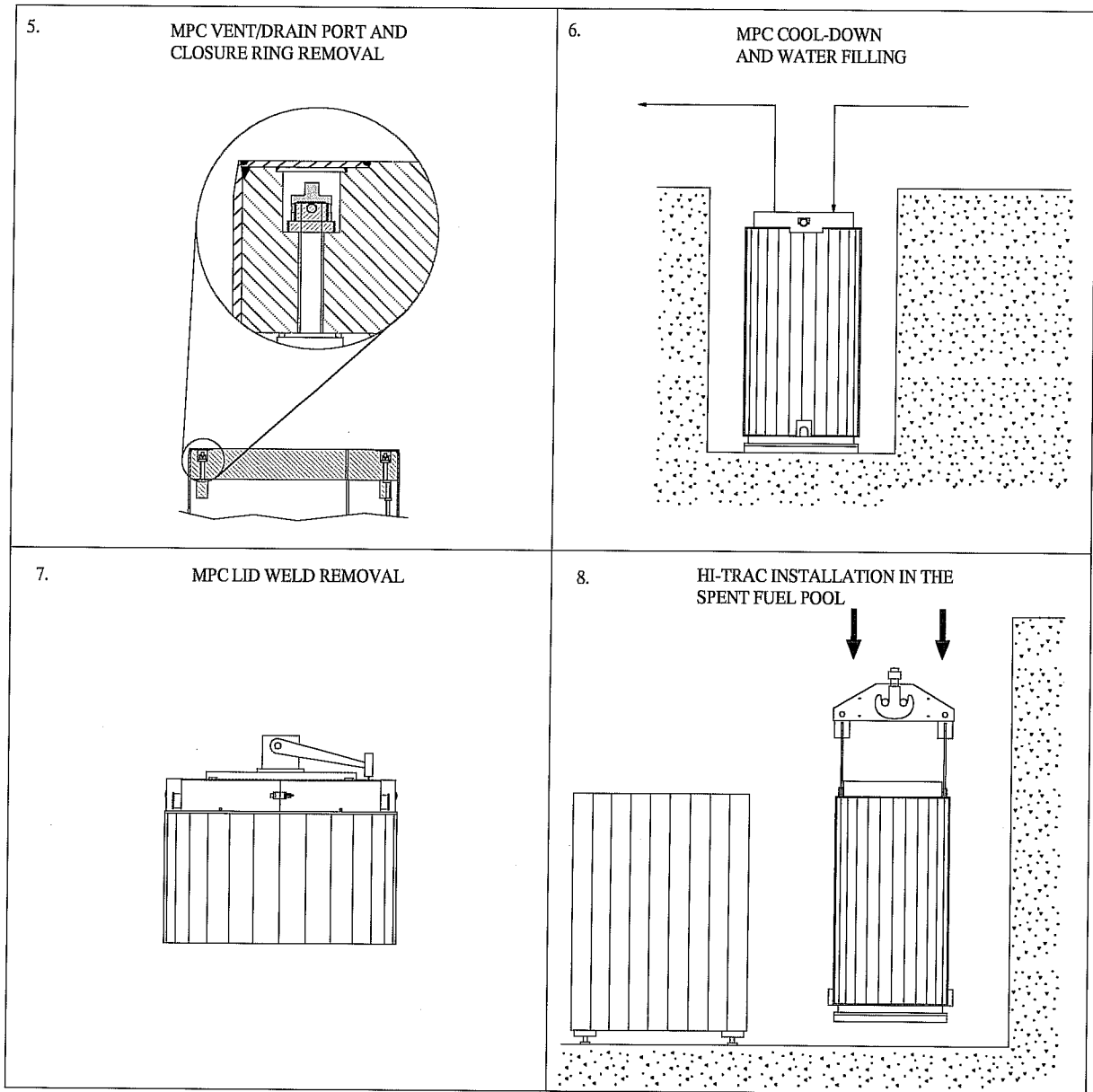
3. BOTTOM LID REPLACEMENT



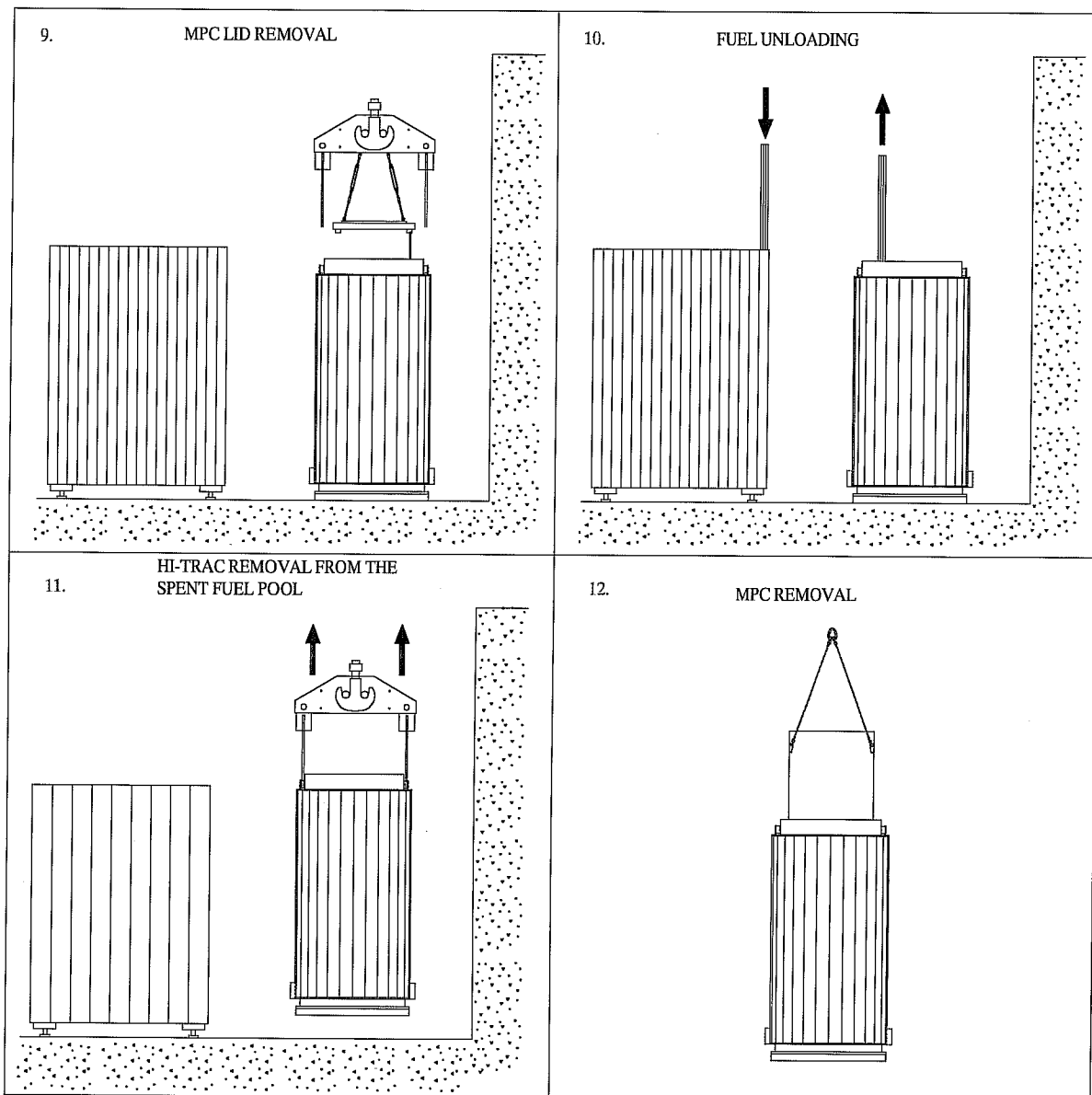
4. HI-TRAC UPPER SHIELD PLATE REMOVAL



**Figure 1.2.17a; Major HI-STORM 100 Unloading Operations (Sheet 1 of 4)**



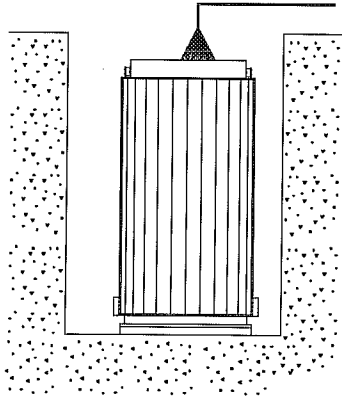
**Figure 1.2.17b; Major HI-STORM 100 Unloading Operations (Sheet 2 of 4)**



**Figure 1.2.17c; Major HI-STORM 100 Unloading Operations (Sheet 3 of 4)**

13.

MPC AND HI-TRAC DECONTAMINATION



**Figure 1.2.17d; Major HI-STORM 100 Unloading Operations (Sheet 4 of 4)**

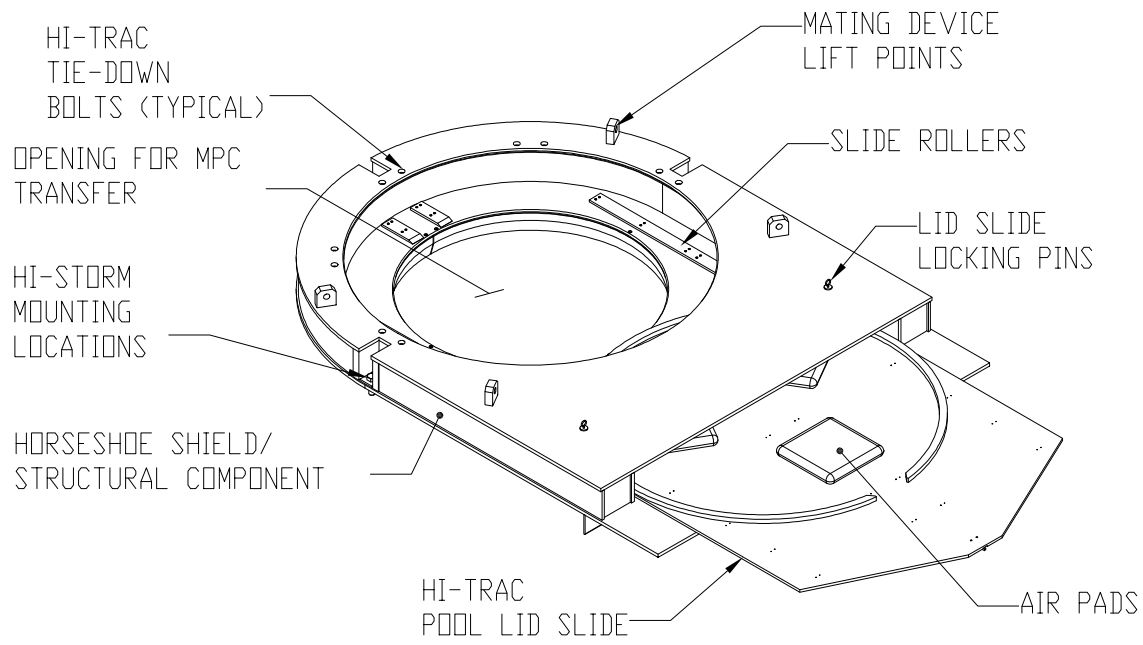


FIGURE 1.2.18; HI-STORM MATING DEVICE



### 1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

Holtec International is a specialty engineering company with a principal focus on spent fuel storage technologies. Holtec has carried out turnkey wet storage capacity expansions (engineering, licensing, fabrication, removal of existing racks, performance of underwater modifications, volume reduction of the old racks and hardware, installation of new racks, and commissioning of the pool for increased storage capacity) in numerous plants around the world. Over 45 plants in the U.S., Britain, Brazil, Korea, and Taiwan have utilized Holtec's wet storage technology to extend their in-pool storage capacity.

Holtec's corporate engineering consists of experts with advanced degrees (Ph.D.'s) in every discipline germane to the fuel storage technologies, namely structural mechanics, heat transfer, computational fluid dynamics, and nuclear physics. All engineering analyses for Holtec's fuel storage projects (including HI-STORM 100) are carried out in-house.

Holtec International's quality assurance program was originally developed to meet NRC requirements delineated in 10CFR50, Appendix B, and was expanded to include provisions of 10CFR71, Subpart H, and 10CFR72, Subpart G, for structures, systems, and components designated as important to safety. ~~A description of the quality assurance program and its method of~~ *The Holtec quality assurance program, which satisfies* ~~satisfying~~ all 18 criteria in 10CFR72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of structures, systems, and components important to safety is *incorporated by reference into this FSAR as described* ~~provided~~ in Chapter 13.

It is currently planned that the HI-STORM 100 System will be fabricated by U.S. Tool & Die, Inc. (UST&D) of Pittsburgh, Pennsylvania. UST&D is an N-Stamp holder and a highly respected fabricator of nuclear components. UST&D is on Holtec's Approved Vendors List (AVL) and has a quality assurance program meeting 10CFR50 Appendix B criteria. Extensive prototypical fabrication of the MPCs has been carried out at the UST&D shop to resolve fixturing and tolerance issues. If another fabricator is to be used for the fabrication of any part of the HI-STORM 100 System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's quality assurance program. ~~described in Chapter 13.~~

Construction, assembly, and operations on-site may be performed by Holtec or a licensee as the prime contractor. A licensee shall be suitably qualified and experienced to perform selected activities. Typical licensees are technically qualified and experienced in commercial nuclear power plant construction and operation activities under a quality assurance program meeting 10CFR50 Appendix B criteria.

## 1.4 GENERIC CASK ARRAYS

The HI-STORM 100 System is stored in a vertical configuration. The required center-to-center spacing between the modules (layout pitch) is guided by operational considerations. Tables 1.4.1 and 1.4.2 provide the nominal layout pitch information. Site-specific pitches are determined by practical operation with supporting heat transfer calculations in Chapter 4. The pitch values in Tables 1.4.1 and 1.4.2 are nominal and may be varied to suit the user's specific needs.

Table 1.4.1 provides recommended cask spacing data for array(s) of two by N casks. The pitch between adjacent rows of casks and between each adjacent column of casks are denoted by  $P_1$  and  $P_2$  in Table 1.4.1. There may be an unlimited number of rows. The distance between adjacent arrays of two by N casks ( $P_3$ ) shall be as specified in Table 1.4.1. See Figure 1.4.1 for further clarification. The pattern of required pitches and distances may be repeated for an unlimited number of columns.

For a square array of casks the pitch between adjacent casks may be in accordance with Table 1.4.2. See Figure 1.4.2 for further clarification. The data in Table 1.4.2 provide nominal values for large ISFSIs (i.e., those with hundreds of casks in a uniform layout), where access of feed air to the centrally located casks may become a matter of thermal consideration. From a thermal standpoint, regardless of the size of the ISFSI, the casks should be arrayed in such a manner that the tributary area for each cask (open ISFSI area attributable to a cask) is a minimum of 225 ft<sup>2</sup>. Subsection 4.4.1.1.7 provides the detailed thermal evaluation of the required tributary area. For specific sites, a smaller tributary area can be utilized after appropriate thermal evaluations for the site-specific conditions are performed.

Table 1.4.1

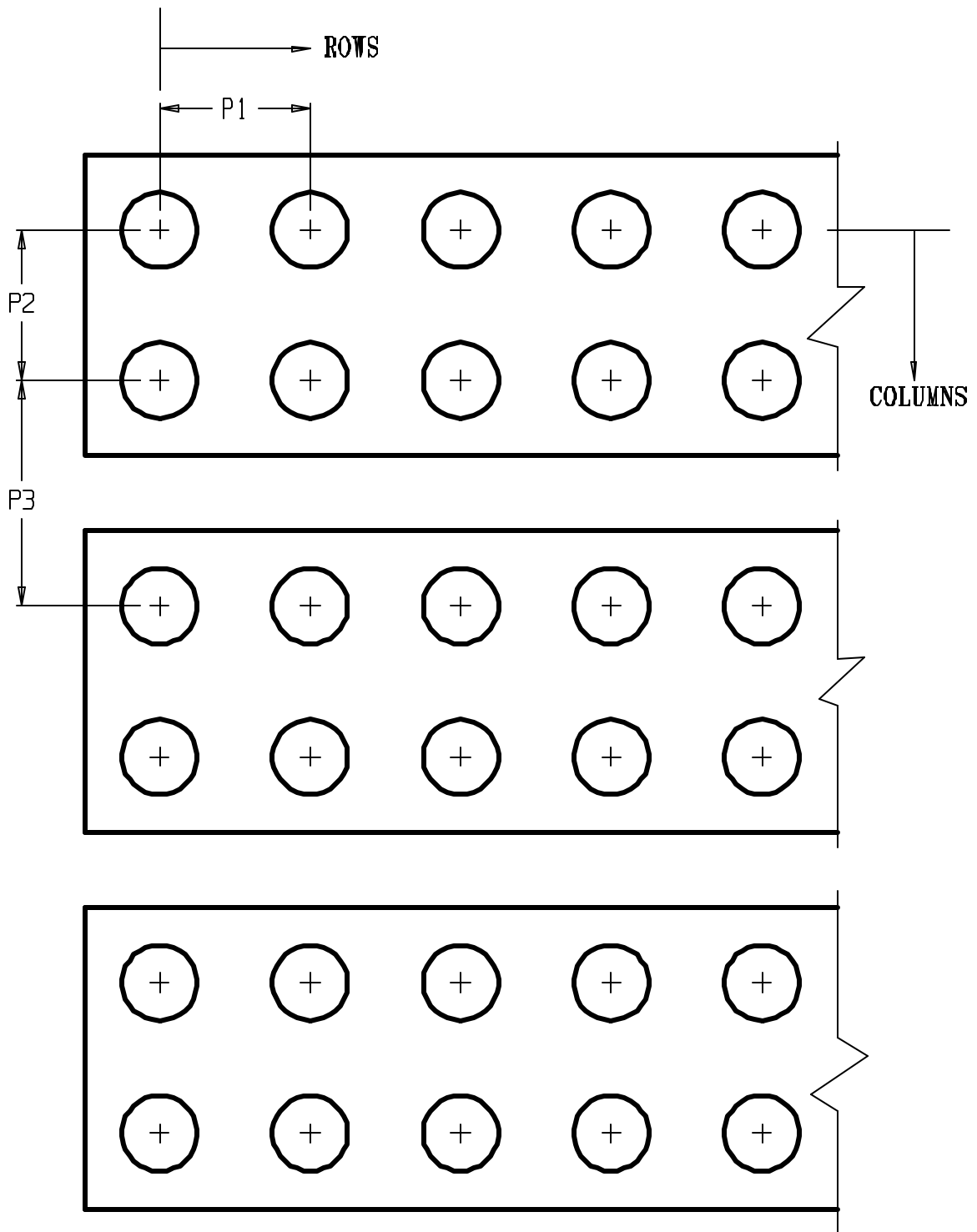
CASK LAYOUT PITCH DATA FOR 2 BY N ARRAYS

<b>Orientation</b>	<b>Nominal Cask Pitch (ft.)</b>
Between adjacent rows, P1, and adjacent columns, P2	13.5
Between adjacent sets of two columns, P3	38

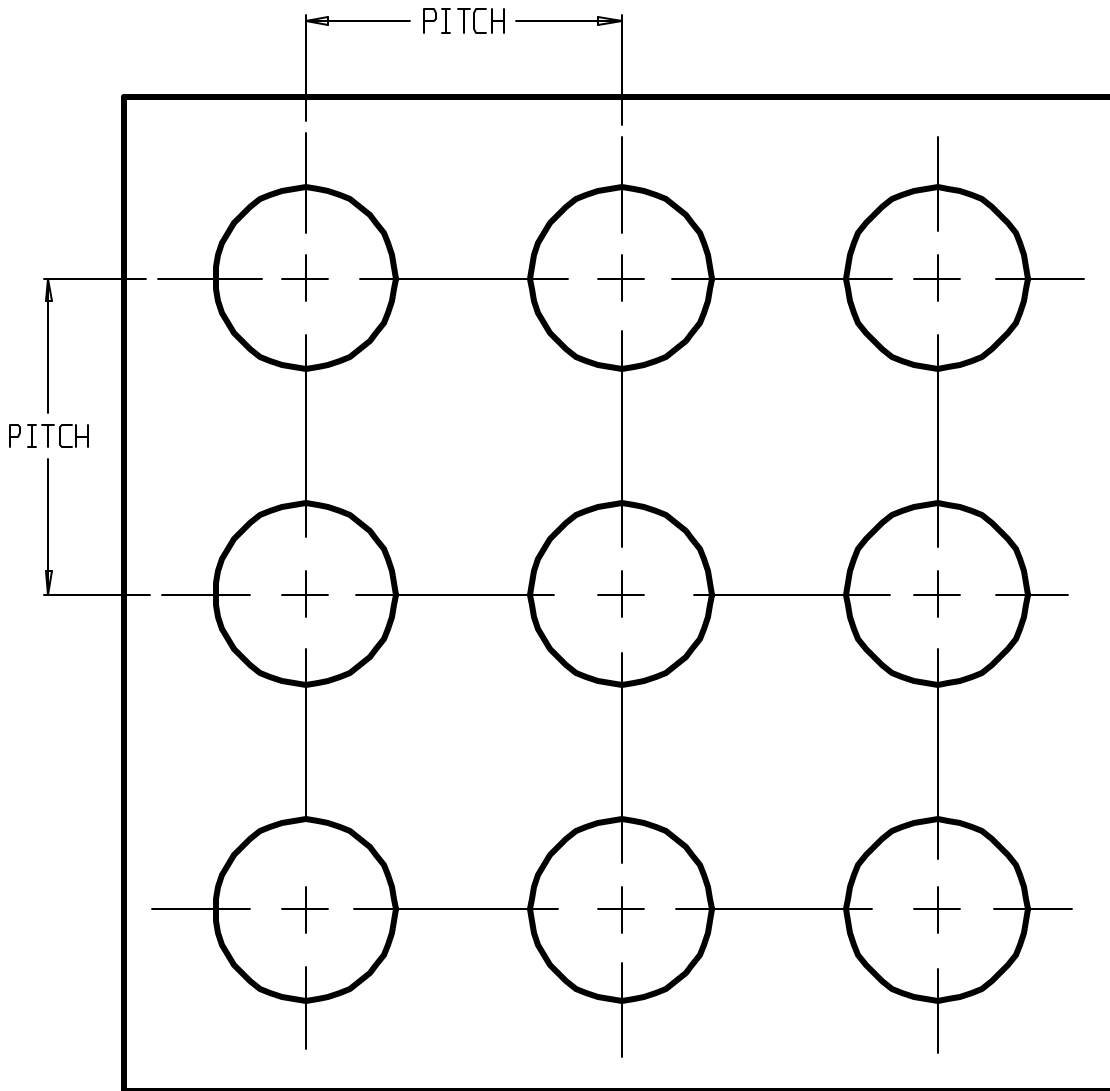
Table 1.4.2

CASK LAYOUT PITCH DATA FOR SQUARE ARRAYS

<b>Orientation</b>	<b>Nominal Cask Pitch (ft.)</b>
Between adjacent casks	18' - 8"



**FIGURE 1.4.1; CASK LAYOUT PITCH REQUIREMENTS  
BASED ON 2 BY N ARRAY(S)**



**FIGURE 1.4.2; CASK LAYOUT PITCH REQUIREMENTS  
BASED ON A SQUARE ARRAY**

## 1.5 GENERAL ARRANGEMENT DRAWINGS

The following HI-STORM 100 System design drawings and bills of materials are provided on subsequent pages in this subsection:

<b>Drawing Number/Sheet</b>	<b>Description</b>	<b>Rev.</b>
3923	MPC Enclosure Vessel	2
3925	MPC-24E/EF Fuel Basket Assembly	1
3926	MPC-24 Fuel Basket Assembly	1
3927	MPC-32 Fuel Basket Assembly	2
3928	MPC-68/68F/68FF Basket Assembly	1
1495 Sht 1/6	HI-STORM 100 Assembly	11
1495 Sht 2/6	Cross Section "Z" - "Z" View of HI-STORM	16
1495 Sht 3/6	Section "Y" - "Y" of HI-STORM	12
1495 Sht 4/6	Section "X" - "X" of HI-STORM	12
1495 Sht 5/6	Section "W" - "W" of HI-STORM	14
1561 Sht 1/6	View "A" - "A" of HI-STORM	10
1561 Sht 2/6	Detail "B" of HI-STORM	13
1561 Sht 3/6	Detail of Air Inlet of HI-STORM	11
1561 Sht 4/6	Detail of Air Outlet of HI-STORM	12
3669	HI-STORM 100S Assembly	5
1880 Sht 1/10	125 Ton HI-TRAC Outline with Pool Lid	9
1880 Sht 2/10	125 Ton HI-TRAC Body Sectioned Elevation	10
1880 Sht 3/10	125 Ton HI-TRAC Body Sectioned Elevation "B" - "B"	9
1880 Sht 4/10	125 Ton Transfer Cask Detail of Bottom Flange	10
1880 Sht 5/10	125 Ton Transfer Cask Detail of Pool Lid	10
1880 Sht 6/10	125 Ton Transfer Cask Detail of Top Flange	10
1880 Sht 7/10	125 Ton Transfer Cask Detail of Top Lid	9
1880 Sht 8/10	125 Ton Transfer Cask View "Y" - "Y"	9
1880 Sht 9/10	125 Ton Transfer Cask Lifting Trunnion and Locking Pad	7
1880 Sht 10/10	125 Ton Transfer Cask View "Z" - "Z"	9
1928 Sht 1/2	125 Ton HI-TRAC Transfer Lid Housing Detail	11
1928 Sht 2/2	125 Ton HI-TRAC Transfer Lid Door Detail	10
2145 Sht 1/10	100 Ton HI-TRAC Outline with Pool Lid	8
2145 Sht 2/10	100 Ton HI-TRAC Body Sectioned Elevation	8
2145 Sht 3/10	100 Ton HI-TRAC Body Sectioned Elevation 'B-B'	8
2145 Sht 4/10	100 Ton HI-TRAC Detail of Bottom Flange	7
2145 Sht 5/10	100 Ton HI-TRAC Detail of Pool Lid	6
2145 Sht 6/10	100 Ton HI-TRAC Detail of Top Flange	8
2145 Sht 7/10	100 Ton HI-TRAC Detail of Top Lid	8
2145 Sht 8/10	100 Ton HI-TRAC View Y-Y	8
2145 Sht 9/10	100 Ton HI-TRAC Lifting Trunnions and Locking Pad	5
2145 Sht 10/10	100 Ton HI-TRAC View Z-Z	7
2152 Sht 1/2	100 Ton HI-TRAC Transfer Lid Housing Detail	10
2152 Sht 2/2	100 Ton HI-TRAC Transfer Lid Door Detail	8

<b>Drawing Number/Sheet</b>	<b>Description</b>	<b>Rev.</b>
3187	Lug and Anchoring Detail for HI-STORM 100A	2
BM-1575, Sht 1/2	Bill-of-Materials HI-STORM 100 Storage Overpack	15
BM-1575, Sht 2/2	Bill-of-Materials HI-STORM 100 Storage Overpack	17
BM-1880, Sht 1/2	Bill-of-Material for 125 Ton HI-TRAC	9
BM-1880, Sht 2/2	Bill-of-Material for 125 Ton HI-TRAC	7
BM-1928, Sht 1/1	Bill-of-Material for 125 Ton HI-TRAC Transfer Lid	10
BM-2145 Sht 1/2	Bill-of-Material for 100 Ton HI-TRAC	6
BM-2145 Sht 2/2	Bill-of-Material for 100 Ton HI-TRAC	5
BM-2152 Sht 1/1	Bill-of-Material for 100 Ton HI-TRAC Transfer Lid	8
3768	125 Ton HI-TRAC 125D Assembly	1



## 1.6 REFERENCES

- [1.0.1] 10CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.
- [1.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989.
- [1.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.
- [1.0.4] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan<sup>†</sup>
- [1.0.5] American Concrete Institute, "Building Code Requirements for Structural Concrete", ACI 318-95, ACI, Detroit, Michigan.
- [1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, 1995 with Addenda through 1997.
- [1.1.2] USNRC Docket No. 72-1008, Final Safety Analysis Report for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.
- [1.1.3] USNRC Docket No. 71-9261, Safety Analysis Report for Packaging for the (Holtec International Storage, Transport, and Repository) HI-STAR System, latest revision.
- [1.1.4] 10CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.
- [1.1.5] Deleted.
- [1.2.1] U.S. NRC Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket".
- [1.2.2] Directory of Nuclear Reactors, Vol. II, Research, Test & Experimental Reactors, International Atomic Energy Agency, Vienna, 1959.
- [1.2.3] V.L. McKinney and T. Rockwell III, "Boral: A New Thermal-Neutron Shield", USAEC Report AECD-3625, August 29, 1949.

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<sup>†</sup> The 1997 edition of ACI-349 is specified for ISFSI pad and embedment design for deployment of the anchored HI-STORM 100A and HI-STORM 100SA.

- [1.2.4] Reactor Shielding Design Manual, USAEC Report TID-7004, March 1956.
- [1.2.5] "Safety Analysis Report for the NAC Storable Transport Cask", Revision 8, September 1994, Nuclear Assurance Corporation (USNRC Docket No. 71-9235).
- [1.2.6] Deleted.
- [1.2.7] Materials Handbook, 13<sup>th</sup> Edition, Brady, G.S. and H.R. Clauser, McGraw-Hill, 1991, Page 310.
- [1.2.8] Deleted.
- [1.2.9] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, June, 1993.
- [1.2.10] Deleted.
- [1.2.11] *"Qualification of METAMIC<sup>®</sup> for Spent Fuel Storage Application," EPRI, 1003137, Final Report, October 2001.*
- [1.2.12] *"Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications," Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., Docket No. 50-313 and 50-368, USNRC, June 2003.*
- [1.2.13] *"Metamic 6061+40% Boron Carbide Metal Matrix Composite Test", California Consolidated Tech. Inc. Report dated August 21, 2001 to NAC International.*
- [1.2.14] *"Recommendations for Preparing the Criticality Safety Evaluation for Transportation Packages," NUREG/CR-5661, USNRC, Dyer and Parks, ORNL.*

## APPENDIX 1.A: ALLOY X DESCRIPTION

### 1.A ALLOY X DESCRIPTION

#### 1.A.1 Alloy X Introduction

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

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

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

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

#### 1.A.2 Alloy X Common Material Properties

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

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

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

#### 1.A.3 Alloy X Least Favorable Material Properties

The following material properties vary between the Alloy X constituents:

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

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

1.A.1 through 1.A.5 provide the ASME Code values for each constituent of Alloy X along with the least favorable value utilized in this licensing application. The ASME Code only provides values to -20°F. The design temperature of the MPC is -40°F to 725°F as stated in Table 1.2.3. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STAR 100 System have been assumed to be equal to the lowest value stated in the ASME Code. The lone exception is the thermal conductivity. The thermal conductivity decreases with the decreasing temperature. The thermal conductivity value for -40°F is linearly extrapolated from the 70°F value using the difference from 70°F to 100°F.

The Alloy X material properties are the minimum values of the group for the design stress intensity, tensile strength, yield strength, and coefficient of thermal conductivity. Using minimum values of design stress intensity is conservative because lower design stress intensities lead to lower allowables that are based on design stress intensity. Similarly, using minimum values of tensile strength and yield strength is conservative because lower values of tensile strength and yield strength lead to lower allowables that are based on tensile strength and yield strength. When compared to calculated values, these lower allowables result in factors of safety that are conservative for any of the constituent materials of Alloy X. Further discussion of the justification for using the minimum values of coefficient of thermal conductivity is given in Chapter 3. The maximum and minimum values are used for the coefficient of thermal expansion of Alloy X. The maximum and minimum coefficients of thermal expansion are used as appropriate in this submittal. Figures 1.A.1-1.A.5 provide a graphical representation of the varying material properties with temperature for the Alloy X materials.

#### 1.A.4 References

[1.A.1] ASME Boiler & Pressure Vessel Code Section II, 1995 ed. with Addenda through 1997.

Table 1.A.1

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

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	20.0	20.0	20.0	20.0	20.0
100	20.0	20.0	20.0	20.0	20.0
200	20.0	20.0	20.0	20.0	20.0
300	20.0	20.0	20.0	20.0	20.0
400	18.7	18.7	19.3	18.9	18.7
500	17.5	17.5	18.0	17.5	17.5
600	16.4	16.4	17.0	16.5	16.4
650	16.2	16.2	16.7	16.0	16.0
700	16.0	16.0	16.3	15.6	15.6
750	15.6	15.6	16.1	15.2	15.2
800	15.2	15.2	15.9	14.9	14.9

Notes:

1. Source: Table 2A on pages 314, 318, 326, and 330 of [1.A.1].
2. Units of design stress intensity values are ksi.

Table 1.A.2

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

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	75.0	75.0	75.0	75.0	75.0
100	75.0	75.0	75.0	75.0	75.0
200	71.0	71.0	75.0	75.0	71.0
300	66.0	66.0	73.4	70.9	66.0
400	64.4	64.4	71.8	67.1	64.4
500	63.5	63.5	71.8	64.6	63.5
600	63.5	63.5	71.8	63.1	63.1
650	63.5	63.5	71.8	62.8	62.8
700	63.5	63.5	71.8	62.5	62.5
750	63.1	63.1	71.4	62.2	62.2
800	62.7	62.7	70.9	61.7	61.7

Notes:

1. Source: Table U on pages 437, 439, 441, and 443 of [1.A.1].
2. Units of tensile strength are ksi.

Table 1.A.3

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

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	30.0	30.0	30.0	30.0	30.0
100	30.0	30.0	30.0	30.0	30.0
200	25.0	25.0	25.8	25.5	25.0
300	22.5	22.5	23.3	22.9	22.5
400	20.7	20.7	21.4	21.0	20.7
500	19.4	19.4	19.9	19.4	19.4
600	18.2	18.2	18.8	18.3	18.2
650	17.9	17.9	18.5	17.8	17.8
700	17.7	17.7	18.1	17.3	17.3
750	17.3	17.3	17.8	16.9	16.9
800	16.8	16.8	17.6	16.6	16.6

Notes:

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

Table 1.A.4

ALLOY X AND CONSTITUENT COEFFICIENT OF THERMAL EXPANSION  
vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X Maximum	Alloy X Minimum
-40	8.55	8.54	8.55	8.54
100	8.55	8.54	8.55	8.54
150	8.67	8.64	8.67	8.64
200	8.79	8.76	8.79	8.76
250	8.90	8.88	8.90	8.88
300	9.00	8.97	9.00	8.97
350	9.10	9.11	9.11	9.10
400	9.19	9.21	9.21	9.19
450	9.28	9.32	9.32	9.28
500	9.37	9.42	9.42	9.37
550	9.45	9.50	9.50	9.45
600	9.53	9.60	9.60	9.53
650	9.61	9.69	9.69	9.61
700	9.69	9.76	9.76	9.69
750	9.76	9.81	9.81	9.76
800	9.82	9.90	9.90	9.82

Notes:

1. Source: Table TE-1 on pages 590 and 591 of [1.A.1].
2. Units of coefficient of thermal expansion are in./in.-°F x 10<sup>-6</sup>.



Table 1.A.5

ALLOY X AND CONSTITUENT THERMAL CONDUCTIVITY vs. TEMPERATURE

Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X (minimum of constituent values)
-40	8.23	6.96	6.96
70	8.6	7.7	7.7
100	8.7	7.9	7.9
150	9.0	8.2	8.2
200	9.3	8.4	8.4
250	9.6	8.7	8.7
300	9.8	9.0	9.0
350	10.1	9.2	9.2
400	10.4	9.5	9.5
450	10.6	9.8	9.8
500	10.9	10.0	10.0
550	11.1	10.3	10.3
600	11.3	10.5	10.5
650	11.6	10.7	10.7
700	11.8	11.0	11.0
750	12.0	11.2	11.2
800	12.2	11.5	11.5

Notes:

1. Source: Table TCD on page 606 of [1.A.1].
2. Units of thermal conductivity are Btu/hr-ft-°F.

# DESIGN STRESS INTENSITY VS. TEMPERATURE

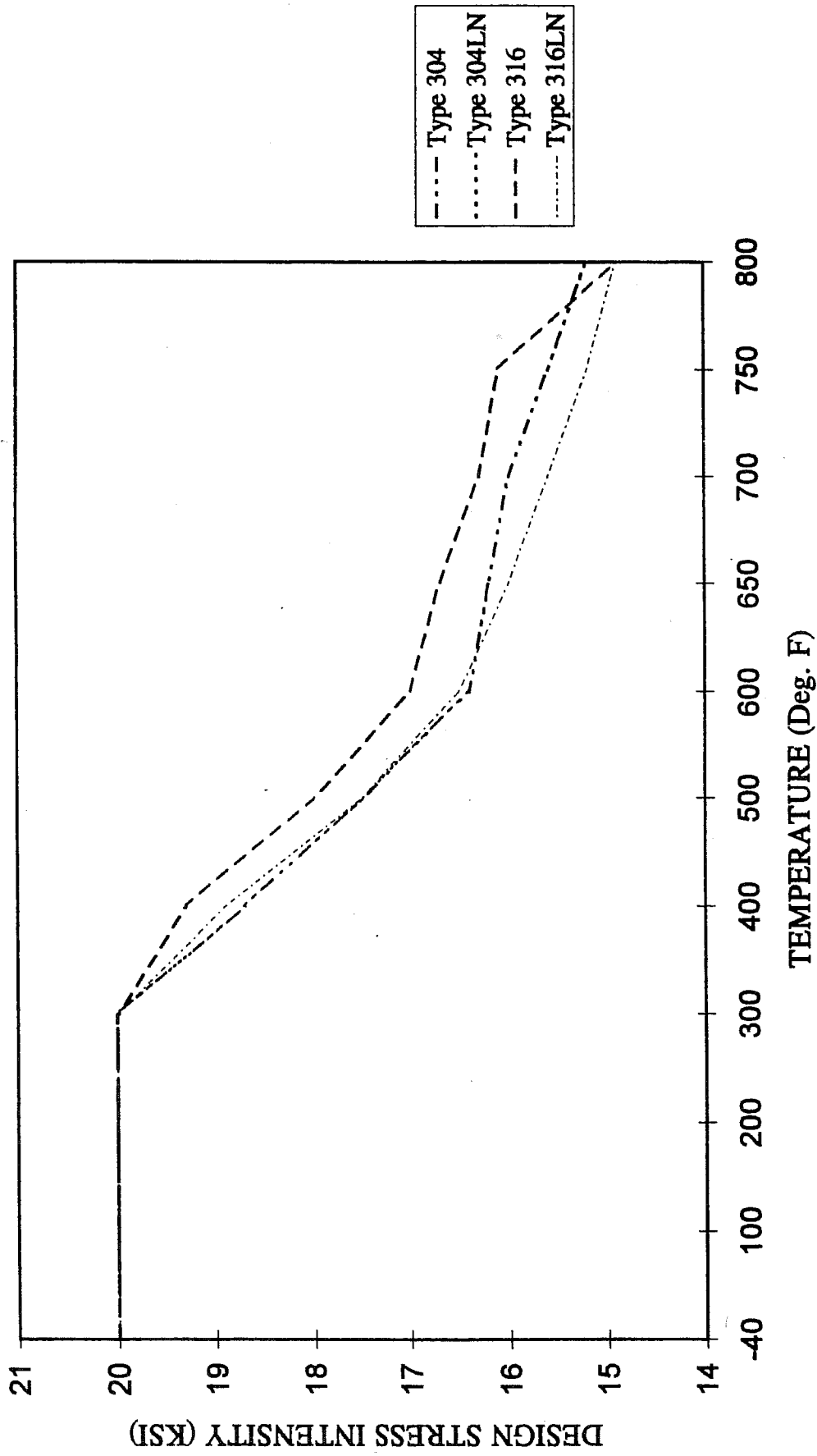


FIGURE 1.A.1; DESIGN STRESS INTENSITY VS. TEMPERATURE

SOURCE: TABLE 1.A.1

# TENSILE STRENGTH VS. TEMPERATURE

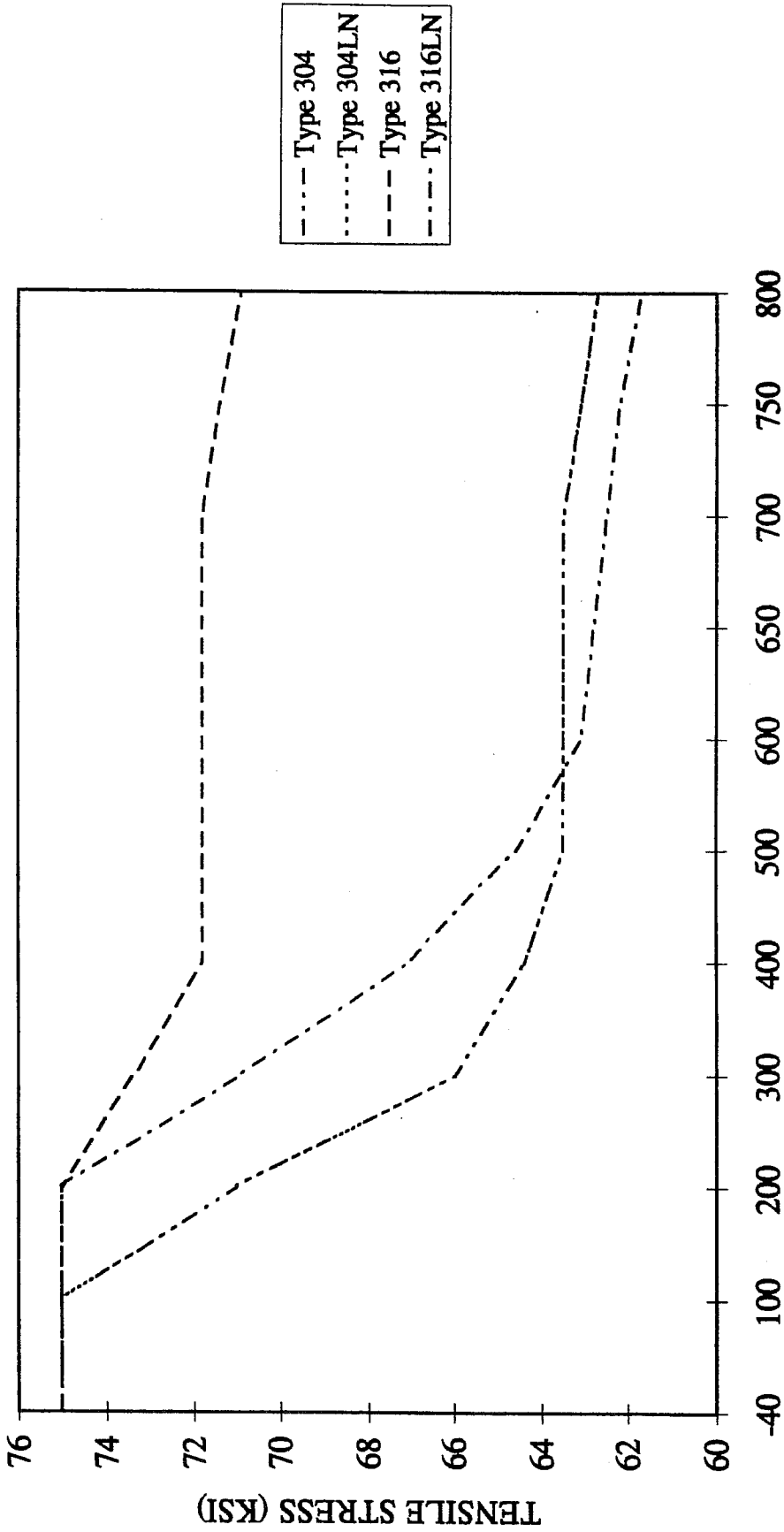


FIGURE 1.A.2; TENSILE STRENGTH VS. TEMPERATURE

SOURCE: TABLE 1.A.2

# YIELD STRESS VS. TEMPERATURE

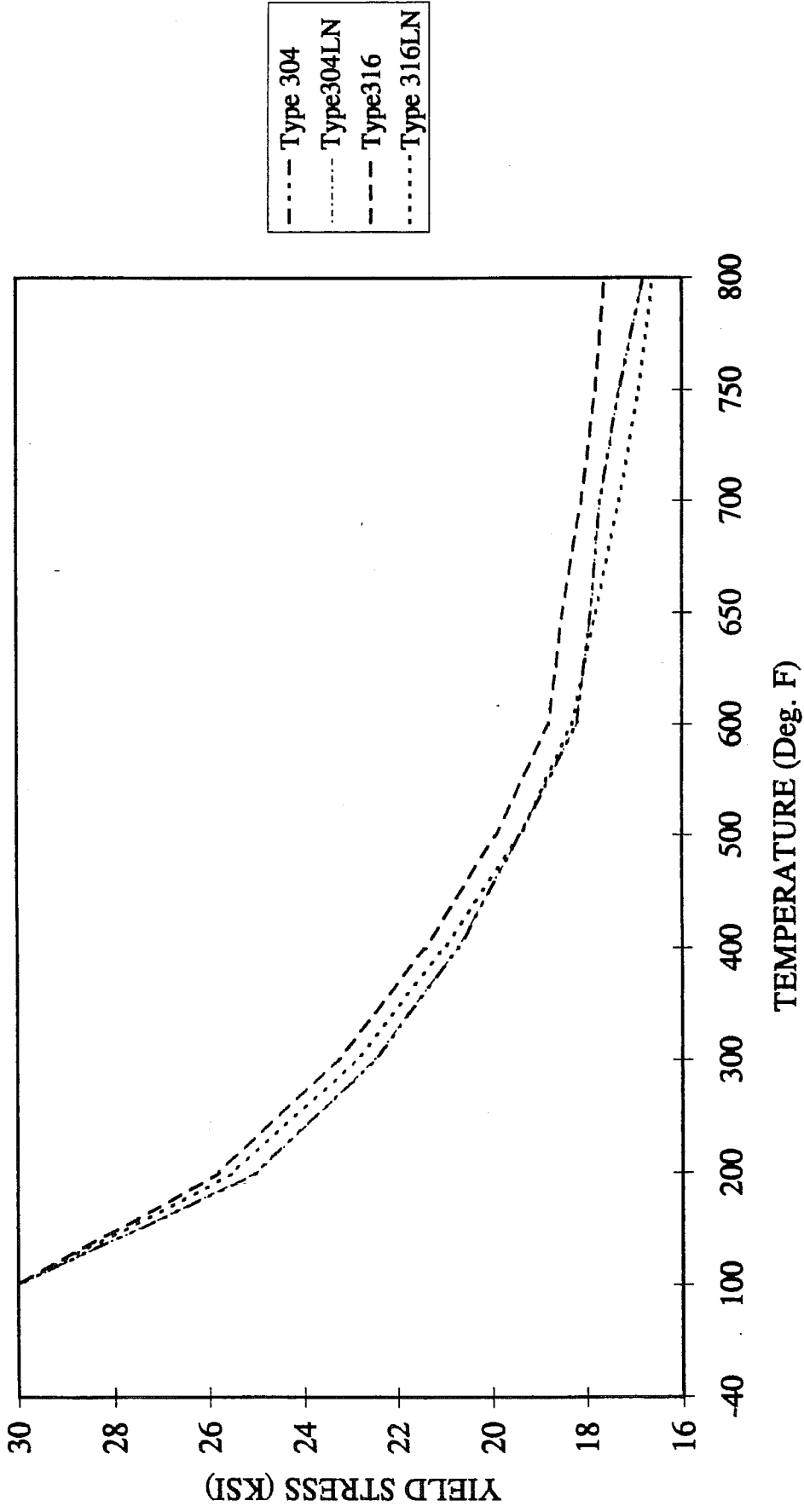
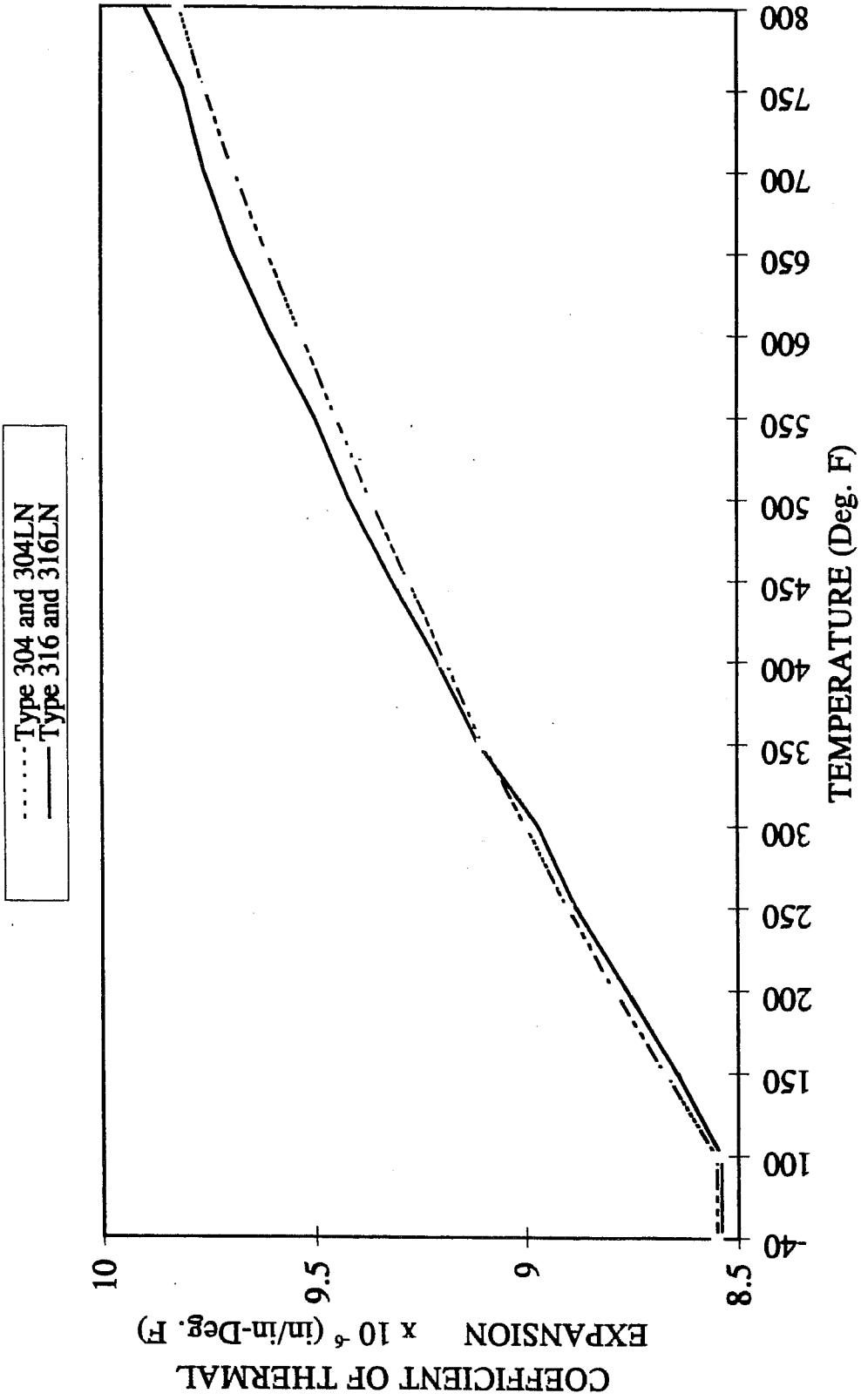


FIGURE 1.A.3; YIELD STRESS VS. TEMPERATURE

SOURCE: TABLE 1.A.3

# COEFFICIENT OF THERMAL EXPANSION VS. TEMPERATURE



SOURCE: TABLE 1.A.4      FIGURE 1.A.4; COEFFICIENT OF THERMAL EXPANSION VS. TEMPERATURE

THERMAL CONDUCTIVITY VS. TEMPERATURE

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

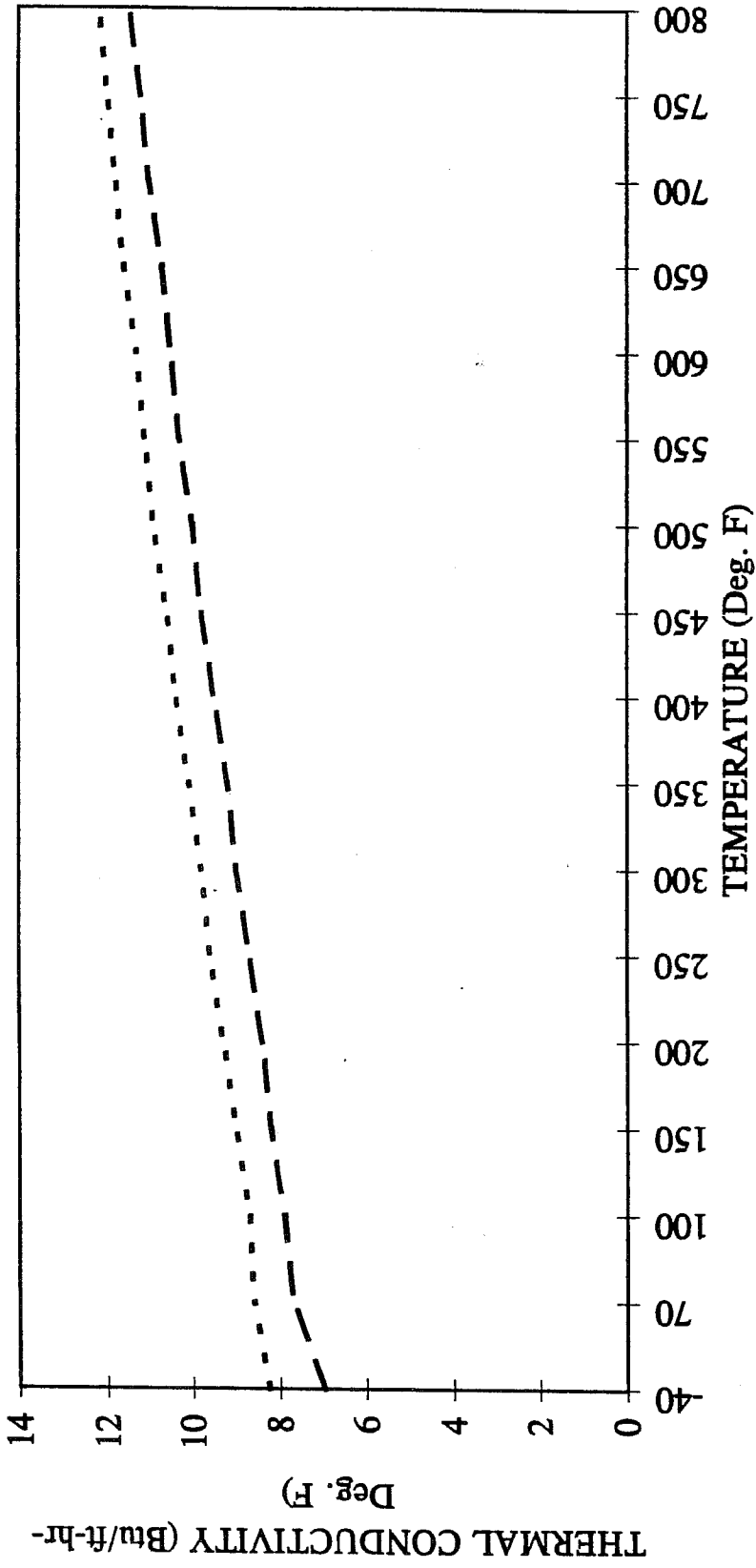


FIGURE 1.A.5; THERMAL CONDUCTIVITY VS. TEMPERATURE

SOURCE: TABLE 1.A.5

## APPENDIX 1.B: HOLTITE™ MATERIAL DATA

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

Holtite-A contains aluminum hydroxide (Al(OH)<sub>3</sub>) in an epoxy resin binder. Aluminum hydroxide is also known by the industrial trade name of aluminum tri-hydrate or ATH. ATH is often used commercially as a fire-retardant. Holtite-A contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A contains 1% (nominal) by weight B<sub>4</sub>C, a chemically inert material added to enhance the neutron absorption property. Pertinent properties of Holtite-A are listed in Table 1.B.1.

The essential properties of Holtite-A are:

1. the hydrogen density (needed to thermalize neutrons),
2. thermal stability of the hydrogen density, and
3. the uniformity in distribution of B<sub>4</sub>C needed to absorb the thermalized neutrons.

ATH and the resin binder contain nearly the same hydrogen density so that the hydrogen density of the mixture is not sensitive to the proportion of ATH and resin in the Holtite-A mixture. B<sub>4</sub>C is added as a finely divided powder and does not settle out during the resin curing process. Once the resin is cured (polymerized), the ATH and B<sub>4</sub>C are physically retained in the hardened resin. Qualification testing for B<sub>4</sub>C throughout a column of Holtite-A has confirmed that the B<sub>4</sub>C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B<sub>4</sub>C is specified in the Holtite-A mixing and pouring procedure as a precaution to assure that the B<sub>4</sub>C concentration is always adequate throughout the mixture.

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

The Holtite-A is encapsulated in the ~~HI-STAR-100 overpack~~ *HI-TRAC transfer cask lid* and, therefore, should experience a very small weight reduction during the design life of the ~~HI-STAR-100 System~~ *cask*. The data and test results confirm that Holtite-A remains stable under design

thermal and radiation conditions, the material properties meet or exceed that assumed in the shielding analysis, and the B<sub>4</sub>C remains uniformly distributed with no evidence of settling or non-uniformity.

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



Table 1.B.1

REFERENCE PROPERTIES OF HOLTITE-A NEUTRON SHIELD MATERIAL

<b>PHYSICAL PROPERTIES</b>	
% ATH	62 nominal
Specific Gravity	1.68 g/cc nominal
Max. Continuous Operating Temperature	300°F
<i>Max. Short-Term Operating Temperature</i>	<i>350°F (Note 1)</i>
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent
<b>CHEMICAL PROPERTIES (Nominal)</b>	
wt% Aluminum	21.5
wt% Hydrogen	6.0
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B <sub>4</sub> C	1.0

NOTES:

1. *As defined in Section 2.2, all operations involving the HI-TRAC transfer cask are short-term operating conditions. The short-term operating temperature limit is, therefore, the appropriate maximum design temperature for the Holtite-A in the HI-TRAC transfer cask.*

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**APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA**  
**(Total of 6 Pages Including This Page)**

The information provided in this appendix specifies the paint properties and demonstrates their suitability for use in spent nuclear fuel storage casks. The following is a listing of the information provided.

- Thermaline 450, Carboline, Product Data Sheet and Application Instructions

Thermaline 450 or equivalent is specified to coat the overpack to the maximum extent practical and the inner cavity of the HI-TRAC transfer cask. Carboline 890 or equivalent is specified to coat external surfaces of the HI-TRAC transfer cask. As can be seen from the product data sheets, the paints are suitable for the design temperatures (see Table 2.2.3) and the environment.

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**THERMALINE 450**



**SELECTION DATA**

**GENERIC TYPE:** A glass flake filled, phenolic modified, amine cured epoxy novalac.

**GENERAL PROPERTIES:** A dense cross-linked polymer which exhibits outstanding barrier protection against a variety of chemical exposures. Excellent resistance to wet/dry cycling conditions at elevated temperatures. Designed to coat the exterior of insulated piping. It is also suitable for coating non-insulated piping and equipment exposed to chemical attack. The glass flakes help provide excellent abrasion resistance, permeation resistance and internal reinforcement.

- Temperature resistance to 450°F
- Excellent abrasion resistance
- Excellent overall chemical resistance
- Excellent thermal shock resistance

**RECOMMENDED USES:** Typically used as a one coat system to coat pipes and tanks that will be insulated. May also be used to coat non-insulated pipe, structural steel, equipment or concrete that may be subjected to severe chemical attack, abrasion or other abuse typical of a chemical plant environment.

**TYPICAL CHEMICAL RESISTANCE:**

<u>Exposure</u>	<u>Splash &amp; Spillage</u>	<u>Fumes</u>
Acids	Excellent	Excellent
Alkalies	Excellent	Excellent
Solvents	Excellent	Excellent
Salt	Excellent	Excellent
Water	Excellent	Excellent

**TEMPERATURE RESISTANCE (Under insulation):**

Continuous: 425°F (218°C)  
Excursions to: 450°F (232°C)

At 200°F (93°C) coating discoloration may be observed without loss of film integrity.

**SUBSTRATES:** Apply over properly prepared steel.

**COMPATIBLE COATINGS:** Normally applied directly to substrate. May be applied over epoxies and phenolics as recommended. May be topcoated with epoxies, polyurethanes or other finish coats as recommended.

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**SPECIFICATION DATA**

**THEORETICAL SOLIDS CONTENT OF MIXED MATERIAL:**

THERMALINE 450 By Volume  
70 ± 2%

**VOLATILE ORGANIC CONTENT (VOC):**

The following are nominal values:  
As supplied: 2.13 lbs./gal. (255 gm./liter).

<u>Thinner</u>	<u>Fluid Ounces/Gal.</u>	<u>Pounds/ Gallon</u>	<u>Grams/ Liter</u>
213	13	2.56	307

**RECOMMENDED DRY FILM THICKNESS:**

8-10 mils (200-250 microns) to be achieved in 1 or 2 coats.

**THEORETICAL COVERAGE PER MIXED GALLON:**

1,117 sq. ft. (27.9 sq.m/l at 25 microns)  
139 sq. ft at 8 mils (3.5 sq. m/l at 200 microns)  
111 sq. ft at 10 mils (2.8 sq.m/l at 250 microns)

\*Mixing and application losses will vary and must be taken into consideration when estimating job requirements.

**STORAGE CONDITIONS:** Store indoors.

Temperature: 40-110°F (4-43°C) Humidity: 0-90%

**SHELF LIFE:** 24 months when stored indoors at 75°F (24°C)

**COLOR:** Red (0500) and Gray (5742)

**GLOSS:** Low (Epoxies lose gloss, discolor and eventually chalk in sunlight exposure.)

**ORDERING INFORMATION**

Prices may be obtained from your Carboline Sales Representative or Carboline Customer Service Department.

**APPROXIMATE SHIPPING WEIGHT:**

	<u>1's</u>	<u>5's</u>
THERMALINE 450	12 lbs. (5.5 kg)	58 lbs. (26.3 kg)
Thinner 213	8.4 lbs. (3.8 kg)	41 lbs. (18.6 kg)

**FLASH POINT: (Setaflash)**

THERMALINE 450 Part A:	53°F	( 12°C)
THERMALINE 450 Part B:	>200°F	(>93°C)
Thinner 213	22°F	( -6°C)

To the best of our knowledge the technical data contained herein are true and accurate at the date of issuance and are subject to change without prior notice. User must contact Carboline Company to verify correctness before specifying or ordering. No guarantee of accuracy is given or implied. We guarantee our products to conform to Carboline quality shown. We assume no responsibility for coverage, performance or injuries resulting from use. Liability, if any, is limited to replacement of products. Prices and cost data shown, are subject to change without prior notice. NO OTHER WARRANTY OR GUARANTEE OF ANY KIND IS MADE BY CARBOLINE, EXPRESS OR IMPLIED. STATUTORY BY OPERATION OF LAW, OR OTHERWISE, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE.

# APPLICATION INSTRUCTIONS

## THERMALINE 450

These instructions are not intended to show product recommendations for specific services. They are issued as an aid in determining correct surface preparation, mixing instructions and application procedure. It is assumed that the proper product recommendations have been made. These instructions should be followed closely to obtain the maximum service from the materials.

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**SURFACE PREPARATION:** Remove all oil or grease from surface to be coated with Thinner 2 or Surface Cleaner 3 (refer to Surface Cleaner 3 instructions) in accordance with SSPC-SP 1.

### STEEL:

**Not Insulated:** Abrasive blast to a Commercial Finish in accordance with SSPC-SP 6 and obtain a 2-3 mil (50-75 micron) blast profile.

**Under Insulation:** Abrasive blast to a Near White Finish in accordance with SSPC-SP 10 and obtain a 2-3 (50-75 micron) blast profile.

**MIXING:** Power mix each component separately, then combine and power mix in the following proportions.

Allow 30 minutes induction time at 75°F (24°C) prior to use.

	<u>1 Gal. Kit</u>	<u>5 Gal. Kit</u>
THERMALINE 450 Part A:	0.8 gals.	4.0 gals.
THERMALINE 450 Part B:	0.2 gals.	1.0 gals.

**THINNING:** May be thinned up to 13 oz/gal with Thinner 213.

Use of thinners other than those supplied or approved by Carboline may adversely affect product performance and void product warranty, whether express or implied.

**POT LIFE:** Three hours at 75°F (24°C) and less at higher temperatures. Pot life ends when coating loses body and begins to sag.

### APPLICATION CONDITIONS:

	<u>Material</u>	<u>Surfaces</u>	<u>Ambient</u>	<u>Humidity</u>
Normal	65-85°F (18-29°C)	65-85°F (18-29°C)	65-85°F (18-29°C)	30-60%
Minimum	55°F (13°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	110°F (43°C)	100°F (38°C)	85%

Do not apply when the surface temperature is less than 5°F or 3°C above the dew point.

Special thinning and application techniques may be required above or below normal conditions.

**SPRAY:** The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

**Conventional:** Pressure pot equipped with dual regulators, 1/2" I.D. minimum material hose, .110" I.D. fluid tip and appropriate air cap.

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**CAUTION:** CONTAINS FLAMMABLE SOLVENTS. KEEP AWAY FROM SPARKS AND OPEN FLAMES. WORKMEN IN CONFINED AREAS MUST WEAR FRESH AIRLINE RESPIRATORS. HYPERSENSITIVE PERSONS SHOULD WEAR GLOVES OR USE PROTECTIVE CREAM. ALL ELECTRICAL EQUIPMENT AND INSTALLATIONS SHOULD BE MADE IN ACCORDANCE WITH THE NATIONAL ELECTRICAL CODE. IN AREAS WHERE EXPLOSION HAZARDS EXIST, WORKMEN SHOULD BE REQUIRED TO USE NONFERROUS TOOLS AND TO WEAR CONDUCTIVE AND NONSPARKING SHOES.

<b>Airless:</b>	
<b>Pump Ratio:</b>	30:1 (min)*
<b>GPM Output:</b>	3.0 (min)
<b>Material Hose:</b>	1/2" I.D. (min)
<b>Tip Size:</b>	.035"-.041"
<b>Output psi:</b>	2200-2500

\*Teflon packings are recommended and are available from the pump manufacturer.

**BRUSH:** For striping of welds, touch-up of small areas only. Use a natural bristle brush, applying full strokes. Avoid rebrushing.

**ROLLER:** Not recommended.

**DRYING TIMES:** These times are based on a dry film thickness of 10 mils (250 microns). Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure.

<u>Surface Temperature</u>	<u>Dry To Handle</u>	<u>Dry to Topcoat</u>	<u>Final Cure</u>
50°F (10°C)	18 hours	48 hours	21 days
60°F (16°C)	12 hours	32 hours	14 days
75°F (24°C)	6 hours	16 hours	7 days
90°F (32°C)	3 hours	8 hours	4 days

If the final cure time has been exceeded, the surface must be abraded by sweep blasting prior to the application of any additional coats.

**EXCESSIVE HUMIDITY OR CONDENSATION ON THE SURFACE DURING CURING MAY RESULT IN A SURFACE HAZE OR BLUSH; ANY HAZE OR BLUSH MUST BE REMOVED BY WATER WASHING BEFORE RE-COATING.**

**VENTILATION & SAFETY: WARNING: VAPORS MAY CAUSE EXPLOSION.** When used in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. In addition to insuring proper ventilation, fresh air respirators or fresh air hoods must be used by all application personnel. Where flammable solvents exist, explosion-proof lighting must be used. Hypersensitive persons should wear clean, protective clothing, gloves and/or protective cream on face, hands and all exposed areas.

**CLEANUP:** Use Thinner 2.

**CAUTION: READ AND FOLLOW ALL CAUTION STATEMENTS ON THIS PRODUCT DATA SHEET AND ON THE MATERIAL SAFETY DATA SHEET FOR THIS PRODUCT.**



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## APPENDIX 1.D: Requirements on HI-STORM 100 Shielding Concrete

### 1.D.1 Introduction

The HI-STORM 100 overpack utilizes plain concrete for neutron and gamma shielding. While most of the shielding concrete used in the HI-STORM 100 overpack is installed in the annulus between the concentric structural shells, smaller quantities of concrete are also present in the pedestal shield and the overpack lid. Because plain concrete has little ability to withstand tensile stresses, but is competent in withstanding compressive and bearing loads, the design of the HI-STORM 100 overpack places no reliance on the tension-competence of the shielding concrete. ACI 318-95 provides formulas for permissible compressive and bearing stresses in plain concrete which incorporate a penalty over the corresponding permissible values in reinforced concrete. The formulas for permissible compressive and bearing stresses set forth in ACI 318-95 are used in calculations supporting this TSAR in load cases involving compression or bearing loads on the overpack concrete. However, since ACI 318-95 is intended for commercial applications and the overpack concrete is designated as an ITS Category B material, it is necessary to invoke provisions of ACI 349 (85) (which is sanctioned by NUREG-1536) for all requirements except for the allowable stress formulas (which do not exist in ACI 349) and load combinations. This appendix provides a complete set of criteria applicable to the plain concrete in the HI-STORM 100 overpack.

### 1.D.2 Design Requirements

The primary function of the plain concrete is to provide neutron and gamma shielding. As plain concrete is a competent structural member in compression, the plain concrete's effect on the performance of the HI-STORM overpack is included. The formulas for permissible compressive and bearing stresses set forth in ACI 318-95 are used. However, as plain concrete has very limited capabilities in tension, no tensile strength is allotted to the concrete.

The steel structure of the HI-STORM overpack provides the strength to meet all load combinations specified in Chapters 2 and 3. Credit for the structural strength of the plain concrete is limited to the compressive load carrying capability of the concrete in calculations appropriate to handling and transfer operations, and to demonstrate that the HI-STORM 100 System continues to provide functional performance in a post-accident environment. Therefore, the load combinations provided in ACI 349 and NUREG-1536, Table 3-1 are not applied to the plain concrete.

The shielding performance of the plain concrete is maintained by ensuring that the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions demonstrate that the plain concrete does not exceed the allowable long term temperature limit provided in Table 1.D.1. Under accident conditions, the bulk of the plain concrete in the HI-STORM overpack does not exceed the allowable short term temperature limit provided in Table 1.D.1. Any portion of the plain concrete which exceeds the short term temperature limit under accident conditions is neglected in the post-accident shielding analysis and in any post-accident structural analysis.



### 1.D.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85) supplemented by the provisions of NUREG 1536 (page 3-21) and standard good practice. Two different minimum concrete densities are specified for the overpack concrete, based on the presence or absence of the steel shield shell.

### 1.D.4 Construction Requirements

The HI-STORM 100 overpack is composed of a steel structure that houses plain concrete. The steel structure acts as the framework for the pouring of the concrete. The steel structure defines the dimensions of the concrete which ensures that the required thickness of concrete is provided. The fabrication sequence for the HI-STORM 100 overpack as it pertains to the concrete is provided below. All item numbers are taken from the design drawings. All nomenclature is taken from the bills-of-material.

The steel structure of the HI-STORM 100 overpack body is assembled at a qualified steel fabrication facility. However, access remains to the annulus formed by the overpack inner and outer shells (Items 3 and 2, respectively); likewise, the pedestal shell (Item 5) is welded to the baseplate (Item 1) and the pedestal platform (Item 24) to form the pedestal cavity, but penetrations exist in the baseplate to allow placement of concrete. The steel structure of the overpack body is transported to the reactor site or a nearby concrete facility.

Once the steel structure of the body is received, the body will be inspected to ensure the steel structure meets the requirements of Sections 5.1 and 6.1 of ACI 349. The concrete shall be mixed, conveyed, and deposited in accordance with Sections 5.2 through 5.4 of ACI 349. Sufficient rigidity in the steel structure overpack body is provided such that all the concrete may be placed in a single pour into each of the four segments formed by the inner shell (Item 3), outer shell (Item 2), and radial plates (Item 14). If more than one pour is performed, the requirements of Section 6.4 of ACI 349 must be met for construction joints. The pedestal shell may require bracing and support in accordance with Section 6.1 of ACI 349 to maintain the proper position and shape.

Mixing and placing of the concrete shall follow the guidance of Sections 5.6 and 5.7 for cold and hot weather conditions, respectively. Consolidation of the plain concrete shall be performed in accordance with ACI 309-87. As no reinforcement is placed in the concrete, the possibility of voids is greatly diminished. Curing of the concrete shall be in accordance with Section 5.5 of ACI 349. Water curing or accelerated curing using sealing materials methods may be used as described in ACI 308-92, Standard Practice for Curing Concrete. This would include the use of either a plastic film or a curing compound.

Non-shrink grout shall be applied as necessary to account for any deviation of the concrete elevation. To fabricate the overpack lid an identical process is followed.

Table 1.D.1 provides the construction limitations and requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

### 1.D.5 Testing Requirements

Table 1.D.2 provides the testing requirements applicable to the overpack plain concrete. These requirements are drawn from ACI 349 (85).

Table 1.D.1: Requirements for Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density in overpack body (Minimum)	146 lb/ft <sup>3</sup> (HI-STORM 100 up to Serial Number (S/N) 7), 155 lb/ft <sup>3</sup> (HI-STORM 100 S/N 8 and higher, and HI-STORM 100S)
Density in lid and pedestal (Minimum)	146 lb/ft <sup>3</sup>
Specified Compressive Strength	3,300 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318-95
Cement Type and Mill Test Report	Type II; Section 3.2 (ASTM C 150 or ASTM C595)
Aggregate Type	Section 3.3 (including ASTM C33( Note 2))
Nominal Maximum Aggregate Size	1 (inch)
Water Quality	Per Section 3.4
Material Testing	Per Section 3.1
Admixtures	Per Section 3.6
Maximum Water to Cement Ratio	0.5 (Table 4.5.2)
Maximum Water Soluble Chloride Ion Cl in Concrete	1.00 percent by weight of cement (Table 4.5.4)
Concrete Quality	Per Chapter 4 of ACI 349
Mixing and Placing	Per Chapter 5 of ACI 349
Consolidation	Per ACI 309-87
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
<del>Maximum</del> <del>Local</del> <del>Through-Thickness Section Average</del> <sup>†</sup> Temperature Limit Under Long Term Conditions	3200°F (See Note 3)
<del>Maximum</del> <del>Through-Thickness Section Average</del> <sup>†</sup> Temperature Limit Under Short Term Conditions	350°F (Appendix A, <del>Subsection-Paragraph A.4.2</del> )
Aggregate Maximum Value <sup>††</sup> of Coefficient of Thermal Expansion (tangent in the range of 70°F to 100°F)	6E-06 inch/inch/°F (NUREG-1536, 3.V.2.b.i.(2)(c)2.b)

<sup>†</sup> The through-thickness section average is the same quantity as that defined in Paragraph A.4.3 of Appendix A to ACI 349 as the mean temperature distribution. A formula for determining this value, consistent with the inner and outer surface averaging used in this FSAR, is presented in Figure A-1 of the commentary on ACI 349. Use of this quantity as an acceptance criterion is, therefore, in accordance with the governing ACI code.

<sup>††</sup> The following aggregate types are a priori acceptable: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite. The thermal expansion coefficient limit does not apply when these aggregates are used. Careful consideration shall be given to the potential of long-term degradation of concrete due to chemical reactions between the aggregate and cement selected for HI-STORM 100 overpack concrete.

*Table 1.D.1 (continued)  
Requirements for Plain Concrete*

Notes:

1. *All section and table references are to ACI 349 (85).*
2. *The coarse aggregate shall meet the requirements of ASTM C33 for class designation 1S from Table 3. However, if the requirements of ASTM C33 cannot be met, concrete that has been shown by special tests or actual service to produce concrete of adequate strength and durability meeting the requirements of Tables 1.D.1 and 1.D.2 is acceptable in accordance with ACI 349 Section 3.3.2.*
3. *The 300°F long term temperature limit is specified in accordance with Paragraph A.4.3 of ACI 349 for normal conditions considering the very low maximum stresses calculated and discussed in Section 3.4 of this FSAR for normal conditions. In accordance with this paragraph of the governing code, the specified concrete compressive strength is supported by test data and the concrete is shown not to deteriorate, as evidenced by a lack of reduction in concrete density or durability. ~~The 200°F long term temperature limit is based on (1) the use of Type II cement, specified aggregate criteria, and the specified compressive stress in Table 1.D.1, (2) the relatively small increase in long term temperature limit over the 150°F specified in Paragraph A.4.1, and (23) the very low maximum stresses calculated for normal and off-normal conditions in Section 3.4 of this FSAR.~~*

Table 1.D.2: Testing Requirements for Plain Concrete

TEST	SPECIFICATION
Compression Test	ASTM C31, ASTM C39, ASTM C192
Unit Weight (Density)	ASTM C138
Maximum Water Soluble Chloride Ion Concentration	Federal Highway Administration Report FHWA-RD-77-85, "Sampling and Testing for Chloride Ion in Concrete"

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

This chapter contains a compilation of design criteria applicable to the HI-STORM 100 System. The loadings and conditions prescribed herein for the MPC, particularly those pertaining to mechanical accidents, are far more severe in most cases than those required for 10CFR72 compliance. The MPC is designed to be in compliance with both 10CFR72 and 10CFR71 and therefore certain design criteria are overly conservative for storage. This chapter sets forth the loading conditions and relevant acceptance criteria; it does not provide results of any analyses. The analyses and results carried out to demonstrate compliance with the design criteria are presented in the subsequent chapters of this report.

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

### 2.0 PRINCIPAL DESIGN CRITERIA

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

#### 2.0.1 MPC Design Criteria

##### General

The MPC is designed for 40 years of service, while satisfying the requirements of 10CFR72. The adequacy of the MPC design for the design life is discussed in Section 3.4.12.

##### Structural

The MPC is classified as important to safety. The MPC structural components include the internal fuel basket and the enclosure vessel. The fuel basket is designed and fabricated as a core support structure, in accordance with the applicable requirements of Section III, Subsection NG of the ASME Code, with certain NRC-approved alternatives, as discussed in Section 2.2.4. The enclosure vessel is designed and fabricated as a Class 1 component pressure vessel in accordance with Section III, Subsection NB of the ASME Code, with certain NRC-approved alternatives, as discussed in Section 2.2.4. The principal exception is the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell, as discussed in Section 2.2.4. In addition, the threaded holes in

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

the MPC lid are designed in accordance with the requirements of ANSI N14.6 for critical lifts to facilitate vertical MPC transfer.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis, as presented in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination of the root pass and/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 hydrostatic pressure test, and a helium leak test, in accordance with the drawings and the CoC.~~

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, hydrostatic pressure testing, and helium leak testing (performed during MPC fabrication) ~~and MPC closure~~, provides assurance of canister closure integrity in lieu of the specific weld joint requirements of 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.

*The structural analysis to evaluate the margin against fuel rod damage from buckling under the drop accident scenario remains unchanged considering ISG-11, Revision 3 because no credit for the tensile stresses in the fuel rods due to internal pressure is taken. Because recognition of the state of tensile axial stress in the fuel cladding permitted by ISG-11 Revision 3 increases the resistance under axial buckling, neglecting the internal pressure buckling analysis is conservative. Therefore, compliance with ISG-11 Revision 3 does not have material effect on the structural analyses summarized in Chapter 3 of this FSAR.*

## Thermal

~~The allowable Zircaloy fuel cladding temperature limits to prevent cladding failure during long term dry storage conditions for moderate burnup fuel in the MPC are based on LLNL Report UCID-21181 [2.2.14]. To provide additional conservatism, the permissible fuel cladding temperature limits, which are lower than those calculated with the LLNL methodology, have been calculated based on PNL Report 6189 [2.0.3]. Stainless steel cladding is demonstrated to withstand higher temperatures than that of Zircaloy cladding in EPRI Report TR-106440 [2.2.13]. However, the Zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel fuel cladding. Allowable fuel cladding temperatures for high burnup fuel assemblies are determined using a creep strain model, developed by Holtec, and described in further detail in Appendix 4.A. The allowable fuel cladding temperatures which correspond to varying cooling times for the SNF to be stored in the MPCs are provided in Table 2.2.3.~~

*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).*
- iv. For High Burnup Fuel (HBF), operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65°C (117°F).*

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

- i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400°C (752°F). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570°C (1058°F) because fuel cladding stress is shown to be less than approximately 90 MPa per Reference [2.0.9]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions added to ensure these limits are met (see Section 4.5).*
- ii. For MPCs containing at least one high burnup fuel (HBF) assembly, the forced helium dehydration (FHD) method of MPC cavity drying must be used to meet the normal operations PCT limit and satisfy the 65°C temperature excursion criterion for HBF.*
- iii. The off-normal and accident condition PCT limit remains unchanged (1058°F).*
- iv. For high burnup fuel, the Supplemental Cooling System (SCS) is required to ensure fuel cladding temperatures remain below the applicable temperature limit (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C.*



The short term allowable fuel cladding temperature that is applicable to off-normal and accident conditions, as well as the fuel loading, canister closure, and canister transfer operations in the HI-TRAC transfer cask, is 570°C (1058°F) based on PNL-4835 [2.2.15]. The MPC cavity is dried using either a vacuum drying system, or a forced helium dehydration system (see Appendix 2.B). The MPC is backfilled with 99.995% pure helium in accordance with the technical specifications limits in Table 1.2.2 during canister sealing operations to promote heat transfer and prevent cladding degradation.

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

The MPCs are designed for a bounding thermal source term, as described in Section 2.1.6. The maximum allowable fuel assembly heat load for each MPC is limited in accordance with the Approved Contents section of Appendix B to the CoC as specified in Section 2.1.9.

Each MPC model, *except MPC-68F*, allows for two fuel loading strategies. The first is uniform fuel loading, wherein any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as preferential fuel loading and location requirements for damaged fuel containers (DFCs) and fuel with integral non-fuel hardware (e.g., control rod assemblies). The second is regionalized fuel loading, wherein the basket is segregated into two regions, as defined in Appendix B to the CoC. Region 1 is the inner region where fuel assemblies with higher heat emission rates may be stored and Region 2 is the outer region where fuel assemblies with lower heat emission rates are stored. Regionalized loading allows for storage of fuel assemblies with higher heat emission rates (in Region 1) than would otherwise be authorized for loading under a uniform loading strategy. Regionalized loading strategies must also comply with other requirements of the CoC, such as those for DFCs and non-fuel hardware. Specific fuel assembly cooling time, burnup, and decay heat limits for regionalized loading are presented in Section 2.1.9 provided in the Approved Contents section of Appendix B to the CoC. The two fuel loading regions are defined by fuel storage location number in Table 2.1.13 (refer to Figures 1.2.2 through 1.2.4). Regionalized fuel loading meets the intent of preferential fuel loading. For MPC-68F, only uniform loading is permitted.

### Shielding

The allowable doses for an ISFSI using the HI-STORM 100 System are delineated in 10CFR72.104 and 72.106. Compliance with these regulations for any particular array of casks at an ISFSI is necessarily site-specific and is to be demonstrated by the licensee, as discussed in Chapters 5 and 12. Compliance with these regulations for a single cask and several representative cask arrays is demonstrated in Chapters 5 and 7.10.

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

The MPCs are designed for design basis fuel as described in Sections 2.1.7 and 5.2. The radiological source term for the MPCs is limited based on the burnup and cooling times specified in ~~Appendix B to the CoC~~ *Section 2.1.9*. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation ~~in accordance with 10CFR20~~, as discussed in Chapter 10.

### Criticality

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

Criticality control is maintained by the geometric spacing of the fuel assemblies, fixed borated neutron absorbing materials (~~Boral~~) incorporated into the fuel basket assembly, and, for certain MPC models, soluble boron in the MPC water. The minimum specified boron concentration verified during ~~Boral~~ *neutron absorber* manufacture is further reduced by 25% for criticality analysis *for Boral-equipped MPCs and by 10% for METAMIC<sup>®</sup>-equipped MPCs*. No credit is taken for burnup. The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC is limited. ~~in accordance with the Approved Contents section of Appendix B to the CoC~~. *Enrichment limits and* soluble boron concentration requirements are delineated in *Section 2.1.9* ~~the Technical Specifications in Appendix A to the CoC~~ *consistent with the criticality analysis described in Chapter 6*.

### Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions, ~~as discussed in Section 7.1~~. *As discussed in Section 7.1, the Holtec MPC design meets the guidance in Interim Staff Guidance 18 to classify confinement boundary leakage as non-credible. A non-mechanistic breach of the canister and subsequent release of available fission products in accordance with specified release fractions is considered, as discussed in Section 7.3. Therefore, no confinement dose analysis is performed.* The confinement function of the MPC is verified through ~~hydrostatic pressure~~ *testing, fabrication shop helium leak testing and weld examinations performed in accordance with the acceptance test program in Chapter 9*.

### Operations

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

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

### Acceptance Tests and Maintenance

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

### Decommissioning

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

## 2.0.2 HI-STORM Overpack Design Criteria

### General

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

### Structural

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

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

Unlike other concrete storage casks, the HI-STORM overpack concrete is enclosed in steel inner and outer shells connected to each other by four radial ribs, and top and bottom plates. Where typical concrete storage casks are reinforced by rebar, the HI-STORM overpack is supported by the inner and outer shells connected by four ribs. As the HI-STORM overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete. Providing further conservatism, the structural analyses for normal conditions demonstrate that the allowable stress limits of the structural steel are met even with no credit for the strength of the concrete. During accident conditions (e.g., tornado missile, tip-over, end drop, and earthquake), only the compressive strength of the concrete is accounted for in the analysis to provide an appropriate simulation of the accident condition. Where applicable, the compressive strength of the concrete is calculated in accordance with ACI-318-95 [2.0.1].

In recognition of the conservative assessment of the HI-STORM overpack concrete strength and the primary function of the concrete being shielding, the applicable requirements of ACI-349 [2.0.2] are invoked in the design and construction of the HI-STORM overpack concrete as specified in Appendix 1.D.

Steel components of the storage overpack are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for Class 3 plate and shell components with certain NRC-approved alternatives.

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the overpack 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 postulated drop accident from the maximum allowable handling height, consistent with the ~~Cask Transport Evaluation program described in Technical Specification Section 5.0 contained in Appendix A to the CoCanalysis described in Section 3.4.9.~~ The load combinations for which the overpack is designed are defined in Section 2.2.7. The physical characteristics of the MPCs for which the overpack is designed are defined in Chapter 1.

### Thermal

The allowable long-term *through-thickness, section-average* temperature limit for the overpack concrete is *established in accordance with Paragraph A.4.3 of Appendix A to ACI 349* ~~less than the limit in NUREG-1536, which allows the use of elevated temperature limits if test data supporting the compressive strength is available and an evaluation to show no concrete deterioration provided.~~ ~~local concrete temperature limit 300°F, if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range.~~ Appendix 1.D specifies the cement and aggregate requirements to allow the utilization of the 300°F temperature limit. ~~of NUREG-1536; however, a conservative long-term temperature limit of 200°F is applied to the concrete.~~ For short term conditions the *through-thickness section average* concrete temperature limit of 350°F is specified in accordance with *Paragraph A.4.2 of Appendix A of* ACI 349. The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in Table 2.2.3.

The overpack is designed for extreme cold conditions, as discussed in Section 2.2.2.2. The structural steel materials used for the storage cask that are susceptible to brittle fracture are discussed in Section 3.1.2.3.

The overpack is designed for the maximum allowable heat load for steady-state normal conditions, in accordance with Section 2.1.6. The thermal characteristics of the MPCs for which the overpack is designed are defined in Chapter 4.

## Shielding

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

The overpack is designed to limit the calculated surface dose rates on the cask for all MPCs as defined in Section 2.3.5. The overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The calculated overpack dose rates are determined in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC transfer operations and a dose assessment for a typical ISFSI, as described in Chapter 10. ~~In addition, overpack dose rates are limited in accordance with the Technical Specifications provided in Appendix A to the CoC.~~

## Confinement

The overpack does not perform any confinement function. Confinement during storage is provided by the MPC and is addressed in Chapter 7. The overpack provides physical protection and biological shielding for the MPC confinement boundary during MPC dry storage operations.

## Operations

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

Generic operating procedures for the HI-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 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 MPC for which the HI-TRAC is designed are defined in Chapter 1.

#### Thermal

The allowable temperatures for the HI-TRAC transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The top lid of the HI-TRAC 100 and HI-TRAC 125 incorporate Holtite-A shielding material. This material has a maximum allowable temperature in accordance with the manufacturer's test data. The specific allowable temperatures for the structural steel and shielding components of the HI-TRAC are provided in Table 2.2.3. The HI-TRAC is designed for off-normal environmental cold conditions, as discussed in Section 2.2.2.2. The structural steel materials susceptible to brittle fracture are discussed in Section 3.1.2.3.

The HI-TRAC is designed for the maximum allowable heat load *analyzed for storage operations*. ~~provided in the technical specifications~~ *When the MPC contains any high burnup fuel assemblies, the Supplemental Cooling System (SCS) will be required for certain time periods while the MPC is inside the HI-TRAC transfer cask (see Section 4.5). The design criteria for the SCS are provided in Appendix 2.C.* The HI-TRAC water jacket maximum allowable temperature is a function of the

internal pressure. To preclude over pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is limited to less than the saturation temperature at the shell design pressure. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the minimum allowable temperature and adding ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1.6. The working area ambient temperature limit for loading operations is *limited in accordance with the design criteria established for the transfer cask, delineated in the Design Features section of Appendix B to the CoC.*

### Shielding

The HI-TRAC transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10CFR20, while also maintaining the maximum load on the plant's crane hook to below either 125 tons or 100 tons, or less, depending on whether the 125-ton or 100-ton HI-TRAC transfer cask is utilized. The HI-TRAC calculated dose rates are reported in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations, as described in Chapter 10. A postulated HI-TRAC accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Section 5.1.2. In addition, HI-TRAC dose rates are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

The HI-TRAC 125 and 125D provide better shielding than the 100 ton HI-TRAC. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would normally dictate that the 125 ton HI-TRAC should be used. However, sites may not be capable of utilizing the 125 ton HI-TRAC due to crane capacity limitations, floor loading limits, or other site-specific considerations. As with other dose reduction-based plant activities, individual users who cannot accommodate the 125 ton HI-TRAC should perform a cost-benefit analysis of the actions (e.g., modifications) which would be necessary to use the 125 ton HI-TRAC. The cost of the action(s) would be weighed against the value of the projected reduction in radiation exposure and a decision made based on each plant's particular ALARA implementation philosophy.

The HI-TRAC provides a means to isolate the annular area between the MPC outer surface and the HI-TRAC inner surface to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC surfaces expected to require decontamination are coated. The maximum permissible surface contamination for the HI-TRAC is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

### Confinement

The HI-TRAC transfer cask does not perform any confinement function. Confinement during MPC transfer operations is provided by the MPC, and is addressed in Chapter 7. The HI-TRAC provides physical protection and biological shielding for the MPC confinement boundary during MPC closure and transfer operations.

### Operations

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

Generic operating procedures for the HI-STORM 100 System are provided in Chapter 8. The licensee will develop detailed operating procedures based on Chapter 8, plant-specific requirements including the Part 50 Technical Specifications, and the HI-STORM 100 System CoC.

### Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the HI-TRAC Transfer Cask are described in Chapter 9. The operational controls and limits to be applied to the HI-TRAC are contained in Chapter 12. Application of these requirements will assure that the HI-TRAC 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 Systems, including the HI-TRAC transfer cask, are addressed in Section 2.4.

#### 2.0.4 Principal Design Criteria for the ISFSI Pad

##### 2.0.4.1 Design and Construction Criteria

In compliance with 10CFR72, Subpart F, “General Design Criteria”, the HI-STORM 100 cask system is classified as “important-to-safety” (ITS). This final safety analysis report (FSAR) explicitly recognizes the HI-STORM 100 System as an assemblage of equipment containing numerous ITS components. The reinforced concrete pad on which the cask is situated, however, is designated as a non-ITS structure. This is principally because, in most cases, cask systems for storing spent nuclear fuel on reinforced concrete pads are installed as free-standing structures. The lack of a physical connection between the cask and the pad permits the latter to be designated as not important-to-safety.

However, if the ZPAs at the surface of an ISFSI pad exceed the threshold limit for free-standing HI-STORM installation set forth in the CoC ~~in this FSAR~~, then the cask must be installed in an anchored configuration (HI-STORM 100A).

In contrast to an ISFSI containing free-standing casks, a constrained-cask installation relies on the structural capacity of the pad to ensure structural safety. The Part 72 regulation require consideration of natural phenomenon in the design. Since an ISFSI pad, in an anchored cask installation, participates in maintaining the stability of the cask during “natural phenomena” on the cask and pad, it is an ITS structure. The procedure suggested in Regulatory Guide 7.10 [2.0.4] and the associated NUREG [2.0.5] indicates that an ISFSI pad used to secure anchored casks should be classified as a Category C ITS structure.



Because tipover of a cask installed in an anchored configuration is not feasible, the pad does not need to be engineered to accommodate this non-mechanistic event. However, the permissible carry height for a loaded HI-STORM 100A overpack must be established for the specific ISFSI pad using the methodology described in this FSAR, if the load handling device is not designed in accordance with ANSI N 14.6 and does not have redundant drop protection design features. These requirements are specified in the CoC. However, to serve as an effective and reliable anchor, the pad must be made appropriately stiff and suitably secured to preclude pad uplift during a seismic event.

Because the geological conditions vary widely across the United States, it is not possible to, a priori, define the detailed design of the pad. Accordingly, in this FSAR, the limiting requirements on the design and installation of the pad are provided. The user of the HI-STORM 100A System bears the responsibility to ensure that all requirements on the pad set forth in this FSAR are fulfilled by the pad design. Specifically, the ISFSI owner must ensure that:

- The pad design complies with the structural provisions of this report. In particular, the requirements of ACI-349-97 [2.0.2] with respect to embedments must be assured.
- The material of construction of the pad (viz., the additives used in the pad concrete), and the attachment system are compatible with the ambient environment at the ISFSI site.
- The pad is designed and constructed in accordance with a Part 72, Subpart G-compliant QA program.
- The design and manufacturing of the cask attachment system are consistent with the provisions of this report.
- Evaluations are performed (e.g., per 72.212) to demonstrate that the seismic and other inertial loadings at the site are enveloped by the respective bounding loadings defined in this report.

A complete listing of design and construction requirements for an ISFSI pad on which an anchored HI-STORM 100A will be deployed is provided in Appendix 2.A. A sample embedment design is depicted in Figure 2.A.1.

#### 2.0.4.2 Applicable Codes

Factored load combinations for ISFSI pad design are provided in NUREG-1536 [2.1.5], which is consistent with ACI-349-85. The factored loads applicable to the pad design consist of dead weight of the cask, thermal gradient loads, impact loads arising from handling and accident events, external missiles, and bounding environmental phenomena (such as earthquakes, wind, tornado, and flood). Codes ACI 360R-92, “Design of Slabs on Grade”; ACI 302.1R, “Guide for Concrete Floor and Slab Construction”; and ACI 224R-90, “Control of Cracking in Concrete Structures” should be used in the design and construction of the concrete pad, as applicable. The embedment design for the HI-STORM 100A (and 100SA) are the responsibility of the ISFSI owner and shall comply with Appendix B to ACI-349-97 as described in Appendix 2.A. A later Code edition may be used

provided a written reconciliation is performed.

The factored load combinations presented in Table 3-1 of NUREG 1536 are reduced in the following to a bounding set of load combinations that are applied to demonstrate adherence to its acceptance criteria.

a. Definitions

- D = dead load including the loading due to pre-stress in the anchor studs
- L = live load
- W = wind load
- $W_t$  = tornado load
- T = thermal load
- F = hydrological load
- E = DBE seismic load
- A = accident load
- H = lateral soil pressure
- $T_a$  = accident thermal load
- $U_c$  = reinforced concrete available strength

Note that in the context of a complete ISFSI design, the DBE seismic load includes both the inertia load on the pad due to its self mass plus the interface loads transmitted to the pad to resist the inertia loads on the cask due to the loaded cask self mass. It is only these interface loads that are provided herein for possible use in the ISFSI structural analyses. The inertia load associated with the seismic excitation of the self mass of the slab needs to be considered in the ISFSI owner's assessment of overall ISFSI system stability in the presence of large uplift, overturning, and sliding forces at the base of the ISFSI pad. Such considerations are site specific and thus beyond the purview of this document.

b. Load Combinations for the Concrete Pad

The notation and acceptance criteria of NUREG-1536 apply.

Normal Events

$$U_c > 1.4D + 1.7L$$

$$U_c > 1.4D + 1.7(L+H)$$

Off-Normal Events

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

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

### Accident-Level Events

$$U_c > D+L+H+T+F$$

$$U_c > D+L+H+T_a$$

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

$$U_c > D+L+H+T+W_t$$

$$U_c > D+L+H+T+A$$

In all of the above load combinations, the loaded cask weight is considered as a live load L on the pad. The structural analyses presented in Chapter 3 provide the interface loads contributing to “E”, “F” and “W<sub>t</sub>”, which, for high-seismic sites, are the most significant loadings. The above set of load combinations can be reduced to a more limited set by recognizing that the thermal loads acting on the ISFSI slab are small because of the low decay heat loads from the cask. In addition, standard construction practices for slabs serve to ensure that extreme fluctuations in environmental temperatures are accommodated without extraordinary design measures. Therefore, all thermal loads are eliminated in the above combinations. Likewise, lateral soil pressure load “H” will also be bounded by “F” (hydrological) and “E” (earthquake) loads. Accident loads “A”, resulting from a tipover, have no significance for an anchored cask. The following three load combinations are therefore deemed sufficient for structural qualification of the ISFSI slab supporting an anchored cask system.

### Normal Events

$$U_c > 1.4D + 1.7(L)$$

### Off-Normal Events

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

### Accident-Level Events

$$U_c > D+L+E \text{ (or } W_t)$$

#### c. Load Combination for the Anchor Studs

The attachment bolts are considered to be governed by the ASME Code, Section III, Subsection NF and Appendix F [2.0.7]. Therefore, applicable load combinations and allowable stress limits for the attachment bolts are as follows:

Event Class and Load Combination	Governing ASME Code Section III Article for Stress Limits
<u>Normal Events</u>	
D	NF-3322.1, 3324.6
<u>Off-Normal Events</u>	
D+F	NF-3322.1, 3324.6 with all stress limits increased by 1.33

Event Class and Load Combination	Governing ASME Code Section III Article for Stress Limits
<u>Accident-Level Events</u>	
D+E and D+W <sub>t</sub>	Appendix F, Section F-1334, 1335

#### 2.0.4.3 Limiting Design Parameters

Since the loaded HI-STORM overpack will be carried over the pad, the permissible lift height for the cask must be determined site-specifically to ensure the integrity of the storage system in the event of a handling accident (uncontrolled lowering of the load). To determine the acceptable lift height, it is necessary to set down the limiting ISFSI design parameters. The limiting design parameters for an anchored cask ISFSI pad and the anchor studs, as applicable, are tabulated in Table 2.0.4. The design of steel embedments in reinforced concrete structures is governed by Appendix B of ACI-349-97. Section B.5 in that appendix states that “anchorage design shall be controlled by the strength of embedment steel...”. Therefore, limits on the strength of embedment steel and on the anchor studs must be set down not only for the purposes of quantifying structural margins for the design basis load combinations, but also for the use of the ISFSI pad designer to establish the appropriate embedment anchorage in the ISFSI pad. The anchored cask pad design parameters presented in Table 2.0.4 allow for a much stiffer pad than the pad for free-standing HI-STORMs (Table 2.2.9). This increased stiffness has the effect of reducing the allowable lift height. However, a lift height for a loaded HI-STORM 100 cask (free-standing or anchored) is not required to be established if the cask is being lifted with a lift device designed in accordance with ANSI N14.6 having redundant drop protection design features.

In summary, the requirements for the ISFSI pad for free-standing and anchored HI-STORM deployment are similar with a few differences. Table 2.0.5 summarizes their commonality and differences in a succinct manner with the basis for the difference fully explained. ~~The CoC provides the specific requirements for ISFSI pad design and establishing lift height limits.~~

#### 2.0.4.4 Anchored Cask/ISFSI Interface

The contact surface between the baseplate of overpack and the top surface of the ISFSI pad defines the structural interface between the HI-STORM overpack and the ISFSI pad. When HI-STORM is deployed in an anchored configuration, the structural interface also includes the surface where the nuts on the anchor studs bear upon the sector lugs on the overpack baseplate. The anchor studs and their fastening arrangements into the ISFSI pad are outside of the structural boundary of the storage cask. While the details of the ISFSI pad design for the anchored configuration, like that for the free-standing geometry, must be custom engineered for each site, certain design and acceptance criteria are specified herein (Appendix 2.A) to ensure that the design and construction of the pad fully comports with the structural requirements of the HI-STORM System.

Table 2.0.1  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Table 1.2.2
License	20 yrs.	10CFR72.42(a) and 10CFR72.236(g)	-
<b>Structural:</b>			
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 <i>for core supports</i> (NG-1121)	10CFR72.24(c)(4)	Section 2.0.1
<i>MPC Fuel Basket Supports</i> (Angled Plates)	<i>ASME Code, Section III, Subsection NG for internal structures (NG-1122)</i>	<i>10CFR72.24(c)(4)</i>	<i>Section 2.0.1</i>
MPC Lifting Points	ANSI N14.6/NUREG-0612	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights <sup>†</sup> :			
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:	<del>100</del> 110 psig	ANSI/ANS 57.9	Section 2.2.2.1
Accident (Internal)	200 psig	ANSI/ANS 57.9	Section 2.2.3.8
Accident (External)	60 psig	ANSI/ANS 57.9	Sections 2.2.3.6 and 2.2.3.10

<sup>†</sup> 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
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(l)	Section 2.0.1
<b>Thermal:</b>			
Maximum Design Temperatures:			
Structural Materials:			
Stainless Steel (Normal)	725° F	ASME Code Section II, Part D	Table 2.2.3
Stainless Steel (Accident)	950° F	ASME Code Section II, Part D	Table 2.2.3
Neutron Poison:			
<del>Boral</del> Neutron Absorber (normal)	800° F	See <del>Section</del> Table 4.3.1 and Section 1.2.1.3.1	Table 2.2.3
<del>Boral</del> Neutron Absorber (accident)	950° F	See <del>Section</del> Table 4.3.1 and Section 1.2.1.3.1	Table 2.2.3
PWR Fuel Cladding (Moderate/High Burnup Fuel):			
5-year cooled	691° / 679° F	<del>PNL 6189/Appendix 4.A</del>	<del>Section 4.3/Appendix 4.A</del>
6-year cooled	676° / 660° F	<del>PNL 6189/Appendix 4.A</del>	<del>Section 4.3/Appendix 4.A</del>
7-year cooled	635° / 635° F	<del>PNL 6189/Appendix 4.A</del>	<del>Section 4.3/Appendix 4.A</del>
10-year cooled	625° / 621° F	<del>PNL 6189/Appendix 4.A</del>	<del>Section 4.3/Appendix 4.A</del>
15-year cooled	614° / 611° F	<del>PNL 6189/Appendix 4.A</del>	<del>Section 4.3/Appendix 4.A</del>
BWR Fuel Cladding (Moderate/High Burnup Fuel):			
5-year cooled	740° F	PNL 6189	<del>Section 4.3/Appendix 4.A</del>
6-year cooled	712° F	PNL 6189	<del>Section 4.3/Appendix 4.A</del>
7-year cooled	669° F	PNL 6189	<del>Section 4. /Appendix 4.A 3</del>
10-year cooled	658° F	PNL 6189	<del>Section 4.3/Appendix 4.A</del>
15-year cooled	646° F	PNL 6189	<del>Section 4.3/Appendix 4.A</del>

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Canister Drying	$\leq 3$ torr for $\geq 30$ minutes (VDS)  $\leq 21^{\circ}\text{F}$ exiting the demohsturizer for $\geq 30$ minutes or a dew point of the MPC exit gas $\leq 22.9^{\circ}\text{F}$ for $\geq 30$ minutes(FHD)	NUREG-1536, ISG-11, Rev. 3	Section 4.5, Appendix 2.B
Canister Backfill Gas	Helium	-	Section <del>4.3</del> 4.4
Canister Backfill	Varies (see Table 1.2.2)	Thermal Analysis	Section 4.34
Fuel cladding temperature limit for long term storage conditions	752 °F (400 °C)	ISG-11, Rev. 3	Section 4.3
Fuel cladding temperature limit for normal short-term operating conditions (e.g., MPC drying and onsite transport)	752 °F (400 °C), except certain MPCs containing all moderate burnup fuel (MBF) may use 1058°F (570°C) for normal short-term operating conditions	ISG-11, Rev. 3	Sections 4.3 and 4.5
<del>Short Term Allowable Fuel</del> Cladding Temperature limit for Off-Normal and Accident Events	1058° F (570 °C)	<del>PNL-4835</del> ISG-11, Rev. 3	Sections 2.0.1 and 4.3
Insolation	Protected by overpack or HI-TRAC	-	Section 4.3
<b>Confinement:</b>		10CFR72.128(a)(3) and 10CFR72.236(d) and (e)	
Closure Welds:			
Shell Seams and Shell-to-Baseplate	Full Penetration	-	Section 1.5 and Table 9.1.4
MPC Lid	Multi-pass Partial Penetration	10CFR72.236(e)	Section 1.5 and Table 9.1.4
MPC Closure Ring	Partial Penetration		
Port Covers	Partial Penetration		

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
<b>NDE:</b>			
Shell Seams and Shell-to-Baseplate	100% RT or UT	-	Table 9.1.4
MPC Lid	Root Pass and Final Surface 100% PT; Volumetric Inspection or 100% Surface PT each 3/8" of weld depth	-	Chapter 8 and Table 9.1.4
Closure Ring	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
Port Covers	Root Pass (if more than one pass is required) and Final Surface 100% PT	-	Chapter 8 and Table 9.1.4
<b>Leak Testing:</b>			
Welds Tested	Shell seams, shell-to-baseplate, MPC lid to shell, and port covers to MPC lid	ISG-18	Section 9.17.1 and Chapters 8, 9, and 12
Medium	Helium	-	Section 9.17.2 and Chapter 12
Max. Leak Rate	$5 \times 10^{-6}$ atm-cm <sup>3</sup> /sec (helium)	-	Section 9.1 Chapter 12 (TS)
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
<b>Hydrostatic Pressure Testing:</b>			
Minimum Test Pressure	125 psig (hydrostatic) 120 psig (pneumatic) (+3, -0 psig)	-	Chapters 8 and 9 Sections 8.1 and 9.1
Welds Tested	MPC Lid-to-Shell, MPC Shell seams, MPC Shell-to-Baseplate	-	Sections 8.1 and 9.1
Medium	Water or helium	-	Section 8.1 and Chapter 9
<b>Retrievability:</b>			
Normal and Off-normal:	No Encroachment on Fuel	10CFR72.122(f),(h)(1), & (l)	Sections 3.4, 3.5, and 3.1.2



Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Post (design basis) Accident	Assemblies or Exceeding Fuel Assembly Deceleration Limits		
<b>Criticality:</b>		10CFR72.124 & 10CFR72.236(c)	
Method of Control	Fixed Borated Neutron Absorber, Geometry, and Soluble Boron	-	Section 2.3.4
Min. <sup>10</sup> Boron Loading (Boral/METAMIC®)	0.0267/0.0223 g/cm <sup>2</sup> (MPC-24) 0.0372/0.0310 g/cm <sup>2</sup> (MPC-68, MPC-68FF, MPC-24E, MPC-24EF, and MPC-32 and MPC-32F) 0.01 g/cm <sup>2</sup> (MPC-68F)	-	Sections 2.1.8 and 6.1
Minimum Soluble Boron	Varies (See Tables 2.1.14 and 2.1.16) CoC, Appendix A, LCO 3.3.1	Criticality Analysis	Sections 2.1.9 and 6.1; CoC, Appendix B
Max. k <sub>eff</sub>	0.95	-	Sections 6.1 and 2.3.4
Min. Burnup	0.0 GWd/MTU (fresh fuel)	-	Section 6.1
<b>Radiation Protection/Shielding:</b>		10CFR72.126, & 10CFR72.128(a)(2)	
MPC: (normal/off-normal/accident)			
MPC Closure	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
MPC Transfer	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
Exterior of Shielding: (normal/off-normal/accident)			
Transfer Mode Position	See Table 2.0.3	10CFR20	Section 5.1.1

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
ISFSI Controlled Area Boundary	See Table 2.0.2	10CFR72.104 & 10CFR72.106	Section 5.1.1 and Chapter 10
<b>Design Bases:</b>		10CFR72.236(a)	
Spent Fuel Specification:			
Assemblies/Canister	Up to 24 (MPC-24, MPC-24E & MPC-24EF) Up to 32 (MPC-32 and MPC-32F) Up to 68 (MPC-68, MPC-68F, & MPC-68FF)	-	Table 1.2.1 and Section 2.1.9
Type of Cladding	<del>Zircaloy</del> ZR and Stainless Steel <sup>*</sup>	-	<del>Table 2.1.6</del> Section 2.1.9
Fuel Condition	Intact, Damaged, and Debris <sup>*</sup>	-	Sections 2.1.2, 2.1.3, and 2.1.9 & Table 2.1.6
<sup>*</sup> See Appendix B to the CoC for specific fuel condition requirements.			
PWR Fuel Assemblies:			
Type/Configuration	Various	-	Section 2.1.9 & Table 2.1.3
Max. Burnup	68,200 MWD/MTU	-	<del>CoC, Appendix B</del> Sections 2.1.9 and 6.2
Max. Enrichment	Varies by fuel design	-	Table 2.1.3 and Section 2.1.9
Max. Decay Heat/Assembly MPC <sup>†</sup> : (Regionalized fuel loading)	28.74 kW	-	Section 4.4
5-year cooled	1470 W (MPC-24) 1540 W (MPC-24E)	—————	Table 4.4.31

<sup>†</sup> Section 2.1.9.1 The Approved Contents Section of Appendix B to the CoC provide describes the decay heat limits per assembly

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	<del>1132 W (MPC 32)</del>		
6-year cooled	<del>1470 W (MPC 24)</del> <del>1540 W (MPC 24E)</del> <del>1072 W (MPC 32)</del>		<del>Table 4.4.31</del>
7-year cooled	<del>1335 W (MPC 24)</del> <del>1395 (MPC 24E)</del> <del>993 (MPC 32)</del>	-	<del>Table 4.4.31</del>
10-year cooled	<del>1235 W (MPC 24)</del> <del>1290 W (MPC 24E)</del> <del>950 W (MPC 32)</del>		<del>Table 4.4.31</del>
15-year cooled	<del>1165 W (MPC 24)</del> <del>1215 W (MPC 24E)</del> <del>918 W (MPC 32)</del>		<del>Table 4.4.31</del>
Minimum Cooling Time:	<del>5</del> 3 years (Intact <del>Zr</del> -Zr Clad Fuel) 8 years (Intact SS Clad Fuel)		<del>CoC, Appendix B</del> Section 2.19
Max. Fuel Assembly Weight: (including non-fuel hardware and DFC, as applicable)	1,680 lb.	-	<del>Table 2.1.6</del> Section 2.1.9
Max. Fuel Assembly Length: (Unirradiated Nominal)	176.8 in.	-	<del>Table 2.1.6</del> Section 2.1.9
Max. Fuel Assembly Width (Unirradiated Nominal)	8.54 in.	-	<del>Table 2.1.6</del> Section 2.1.9
Fuel Rod Fill Gas:			
Pressure (max.)	<del>500</del> psig		<del>Section 4.3 &amp; Table 4.3.2</del>
BWR Fuel Assemblies:			
Type	Various	-	<del>Table 2.1.4</del> Sections 2.1.9 and 6.2
Max. Burnup	<del>59,900</del> 65,000 MWD/MTU	-	<del>CoC, Appendix B</del> Section 2.1.9
Max. Enrichment	Varies by fuel design	-	<del>Section 2.1.9</del> 6.1, Table 2.1.4

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Max. Decay Heat/Assy MPC <sup>†</sup> . (Regionalized Fuel Loading):	28.19 kW	-	Section 4.4
5-year cooled	501 W (MPC 68)	-----	Table 4.4.31
6-year cooled	468 W (MPC 68)	-----	Table 4.4.31
7-year cooled	419 W (MPC 68)	-----	Table 4.4.31
10-year cooled	406 W (MPC 68)	-----	Table 4.4.31
15-year cooled	392 W (MPC 68)	-----	Table 4.4.31
Minimum Cooling Time:	<del>5 yrs</del> 3 years (Intact ZR Clad Fuel) 8 years (Intact SS Clad Fuel)		CoC, Appendix B Section 2.1.9
Max. Fuel Assembly Weight: w/channels and DFC, as applicable	700 lb.	-	Table 2.1.6 Section 2.1.9
Max. Fuel Assembly Length (Unirradiated Nominal)	176.5in.	-	Table 2.1.6 Section 2.1.9
Max. Fuel Assembly Width (Unirradiated Nominal)	5.85 in.	-	Table 2.1.6 Section 2.1.9
Fuel Rod Fill Gas:			
End of Life Hot Standby Pressure (max.)	147 psig	-----	Table 4.3.5
<b>Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperatures	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.1.4
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 Yield	NUREG-0612 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

<sup>†</sup> The Approved Contents Section of Appendix B to the CoC Section 2.1.9.1 provide describes the decay heat limits per assembly.

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
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% (2.5% for high burnup fuel)	NUREG-1536	Sections 2.2.1.3 and 7.2
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3 and 7.2
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3 and 7.2
Snow and Ice	Protected by Overpack	ASCE 7-88	Section 2.2.1.6
<b>Off-Normal Design Event Conditions:</b>		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	<del>No Loss of Confinement</del> N/A	<del>ANSI/ANS 57.9 ISG-18</del>	Sections 2.2.2.4 and 7.1
Partial Blockage of Overpack Air Inlets	Two Air Inlets Blocked	-	Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10% (11.5% for high burnup fuel)	NUREG-1536	Sections 2.2.2.1 and 7.2
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1 and 7.2
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1 and 7.2
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>		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
Side Drop	See Table 2.0.3	-	Section 2.2.3.1
Fire	See Tables 2.0.2 and 2.0.3	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture Releases:			
Fuel Rod Failures (including non-fuel hardware)	100%	NUREG-1536	Sections 2.2.3.8 and 7.3
Fill Gases	100%	NUREG-1536	Sections 2.2.3.8 and 7.3
Fission Gases	30%	NUREG-1536	Sections 2.2.3.8 and 7.3
Particulates & Volatiles	See Table 7.3.1 N/A	-	Sections 2.2.3.9 and 7.3
Confinement Boundary Leakage	<del>None</del> $7.5 \times 10^{-6}$ atm-cm <sup>3</sup> /sec (helium)	TS leak rate plus test sensitivity ISG-18	Sections 2.2.3.9 and 7.1 and 7.3

Table 2.0.1 (continued)  
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Explosive Overpressure	60 psig (external)	10CFR72.122(c)	Section 2.2.3.10
Airflow Blockage:			
Vent Blockage	100% of Overpack Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Partial Blockage of MPC Basket Vent Holes	Crud Depth (Table 2.2.8)	ESEERCO Project EP91-29	Section 2.2.3.4
<b>Design Basis Natural Phenomenon Design Events and Conditions:</b>		10CFR72.92 & 10CFR72.122(b)(2)	
Flood Water Depth	125 ft.	ANSI/ANS 57.9	Section 2.2.3.6
Seismic	See Table 2.0.2	10CFR72.102(f)	Section 2.2.3.7
Wind	Protected by Overpack	ASCE-7-88	Section 2.2.3.5
Tornado & Missiles	Protected by Overpack	RG 1.76 & NUREG-0800	Section 2.2.3.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	See Table 2.0.2	NFPA 78	Section 2.2.3.11
Extreme Environmental Temperature	See Table 2.0.2	-	Section 2.2.3.14

Table 2.0.2  
HI-STORM OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Section 2.0.2
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
<b>Structural:</b>			
Design & Fabrication Codes:			
Concrete			
Design	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Fabrication	ACI 349 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Compressive Strength	ACI 318-95 as specified in Appendix 1.D	10CFR72.24(c)(4)	Section 2.0.2 and Appendix 1.D
Structural Steel			
Design	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Fabrication	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.2
Dead Weights <sup>†</sup> :			
Max. Loaded MPC (Dry)	88,135 lb. (MPC- 32)	R.G. 3.61	Table 3.2.1
Max. Empty Overpack Assembled with Top Lid	270,000 lb.	R.G. 3.61	Table 3.2.1
Max. MPC/Overpack	360,000 lb.	R.G. 3.61	Table 3.2.1
Design Cavity Pressures	N/A	-	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1

<sup>†</sup> 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 2.0.2 (continued)  
 HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
<b>Thermal:</b>			
Maximum Design Temperatures:			
Concrete			
<del>Local Maximum Through-Thickness Section Average</del> (Normal)	2300° F	ACI 349 Appendix A (Paragraph A.4.3)	Section 2.0.2, and Tables 1.D.1 and 2.2.3
<del>Local Maximum Through-Thickness Section Average</del> (Accident)	350° F	ACI 349 Appendix A (Paragraph A.4.2)	Section 2.0.2, and Tables 1.D.1 and -2.2.3
Steel Structure ( <i>other than lid bottom plate</i> ) <i>Lid Bottom Plate</i>	3450° F  400° F	ASME Code Section II, Part D	Table 2.2.3
Insulation:	Averaged Over 24 Hours	10CFR71.71	Section 4.4.1.1.8
<b>Confinement:</b>	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
<b>Retrievability:</b>			
Normal and Off-normal	No damage that precludes Retrieval of MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 and 3.4
Accident			Sections 3.5 and 3.4
<b>Criticality:</b>	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
<b>Radiation Protection/Shielding:</b>		10CFR72.126 & 10CFR72.128(a)(2)	
Overpack (Normal/Off-normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10



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) $\leq$ 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
<b>Design Bases:</b>			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1.9
<b>Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load <sup>†</sup> :		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	245,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.2 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:			
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/6 Yield ANSI N14.6	NUREG-0612 ANSI N14.6	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3

<sup>†</sup> 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 2.0.2 (continued)  
HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature During Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Snow and Ice Load	100 lb./ft <sup>2</sup>	ASCE 7-88	Section 2.2.1.6
Wet/Dry Loading	Dry	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
<b>Off-Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	-40° F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100° F	ANSI/ANS 57.9	Section 2.2.2.2
Partial Blockage of Air Inlets	Two Air Inlet Ducts Blocked	-	Section 2.2.2.5
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>		10CFR72.94	
Drop Cases:			
End	11 in.	-	Section 2.2.3.1
Tip-Over (Not applicable for HI-STORM 100A)	Assumed (Non-mechanistic)	-	Section 2.2.3.2
Fire:			
Duration	217 seconds	10CFR72.122(c)	Section 2.2.3.3
Temperature	1,475° F	10CFR72.122(c)	Section 2.2.3.3
Fuel Rod Rupture	See Table 2.0.1	-	Section 2.2.3.8
Air Flow Blockage:			
Vent Blockage	100% of Air Inlets Blocked	10CFR72.128(a)(4)	Section 2.2.3.13
Ambient Temperature	80° F	10CFR72.128(a)(4)	Section 2.2.3.13
Explosive Overpressure External Differential Pressure	10 psid instantaneous, 5 psid steady state	10 CFR 72.128(a)(4)	Table 2.2.1
<b>Design-Basis Natural Phenomenon Design Events and Conditions:</b>		10CFR72.92 & 10CFR72.122(b)(2)	
Flood			
Height	125 ft.	RG 1.59	Section 2.2.3.6
Velocity	15 ft/sec.	RG 1.59	Section 2.2.3.6

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

Type	Criteria	Basis	FSAR Reference
Seismic			
Max. ZPA-acceleration at top of ISFSI pad	Free Standing: $G_H + 0.53G_V \leq 0.53$ Anchored: $G_H \leq 2.12, G_V \leq 1.5$	10CFR72.102(f)	Section 3.4.7.1 Section 3.4.7.3
Tornado			
Wind			
Max. Wind Speed	360 mph	RG 1.76	Section 2.2.3.5
Pressure Drop	3.0 psi	RG 1.76	Section 2.2.3.5
Missiles			
Automobile			
Weight	1,800 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder			
Weight	125 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	8 in.	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	0.22 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	1 in.	NUREG-0800	Table 2.2.5
Burial Under Debris	Maximum Decay Heat Load	-	Section 2.2.3.12
Lightning	Resistance Heat-Up	NFPA 70 & 78	Section 2.2.3.11
Extreme Environmental Temperature	125° F	-	Section 2.2.3.14
<b>Load Combinations:</b>	See Table 2.2.14 and Table 3.1.5	ANSI/ANS 57.9 and NUREG-1536	Section 2.2.7

TABLE 2.0.3  
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
<b>Design Life:</b>			
Design	40 yrs.	-	Section 2.0.3
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
<b>Structural:</b>			
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	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights <sup>†</sup> :			
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) 143,000 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 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):			
Water Jacket Empty	245,000 lb. (HI-TRAC 125 and 125D) 202,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3
Water Jacket Cavity	60 psig (internal)	ANSI/ANS 57.9	Section 2.2.1.3
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Section 2.0.3

<sup>†</sup> 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
	Continued adequate performance of HI-TRAC transfer cask	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
<b>Thermal:</b>			
Maximum Design Temperature			
Structural Materials	400° F	ASME Code Section II, Part D	Table 2.2.3
Shielding Materials			
Lead	350° F (max.)		Table 2.2.3
Liquid Neutron Shield	307° F (max.)	-	Table 2.2.3
Solid Neutron Shield	300° F (max.) ( <i>long term</i> ) 350° F (max.) ( <i>short term</i> )	Manufacturer Test Data	Appendix 1.B and Table 2.2.3
Insulation:	Averaged Over 24 Hours	10CFR71.71	Section 4.5.1.1.3
<b>Confinement:</b>	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
<b>Retrievability:</b>			
Normal and Off-normal	No encroachment on MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 & 3.4
After Design-basis (Postulated) Accident			Section 3.5 & 3.4
<b>Criticality:</b>	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
<b>Radiation Protection/Shielding:</b>			
Transfer Cask (Normal/Off-normal/Accident)		10CFR72.126 & 10CFR72.128(a)(2)	
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10
<b>Design Bases:</b>			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
<b>Normal Design Event Conditions:</b>			
Ambient Temperatures:			
Lifetime Average	100° F	ANSI/ANS 57.9	Section 2.2.1.4

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

Type	Criteria	Basis	FSAR Reference
Live Load <sup>†</sup>			
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
<b>Handling:</b>			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 Yield	NUREG-0612 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	Section 9.1.2.1
<b>Off-Normal Design Event Conditions:</b>		10CFR72.122(b)(1)	
Ambient Temperature			
Minimum	0° F	ANSI/ANS 57.9	Section 2.2.2.2
Maximum	100° F	ANSI/ANS 57.9	Section 2.2.2.2
<b>Design-Basis (Postulated) Accident Design Events and Conditions:</b>		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1
Fire			
Duration	4.8 minutes	10CFR72.122(c)	Section 2.2.3.3
Temperature	1,475° F	10CFR72.122(c)	Section 2.2.3.3

<sup>†</sup> 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
Fuel Rod Rupture	See Table 2.0.1		Section 2.2.3.8
<b>Design-Basis Natural Phenomenon Design Events and Conditions:</b>		10CFR72.92 & 10CFR72.122(b)(2)	
Missiles			Section 2.2.3.5
Automobile			
Weight	1800 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder			
Weight	125 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	8 in.	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	0.22 kg	NUREG-0800	Table 2.2.5
Velocity	126 mph	NUREG-0800	Table 2.2.5
Diameter	1 in.	NUREG-0800	Table 2.2.5
<b>Load Combinations:</b>	See Table 2.2.14 and Table 3.1.5	ANSI/ANS-57.9 & NUREG-1536	Section 2.2.7

TABLE 2.0.4  
LIMITING DESIGN PARAMETERS FOR ISFSI PADS AND ANCHOR STUDS FOR HI-STORM 100A

Item	Maximum Permitted Value <sup>†</sup>	Minimum Permitted Value
ISFSI PAD		
Pad Thickness	---	48 inches
Subgrade Young's Modulus from Static Tests (needed if pad is not founded on rock)	---	10,000 psi
Concrete compressive strength at 28 days	---	4,000 psi
ANCHOR STUDS		
Yield Strength at Ambient Temperature	None	80,000 psi
Ultimate Strength at Ambient Temperature	None	125,000 psi
Initial Stud Tension	65 ksi	55 ksi

<sup>†</sup> Pad and anchor stud parameters to be determined site-specifically, except where noted.



TABLE 2.0.5  
ISFSI PAD REQUIREMENTS FOR FREE-STANDING AND ANCHORED HI-STORM INSTALLATION

Item	Free-Standing	Anchored	Comments
1. Interface between cask and ISFSI	Contact surface between cask and top surface of ISFSI pad	Same as free-standing with the addition of the bearing surface between the anchor stud nut and the overpack baseplate. (The interface between the anchor stud and the anchor receptacle is at the applicable threaded or bearing surface).	All components below the top surface of the ISFSI pad and in contact with the pad concrete are part of the pad design. A non-integral component such as the anchor stud is not part of the embedment even though it may be put in place when the ISFSI pad is formed. The embedment for the load transfer from the anchor studs to the concrete ISFSI pad shall be exclusively cast-in-place.
2. Applicable ACI Code	At the discretion of the ISFSI owner. ACI-318 and ACI-349 are available candidate codes.	ACI-349-97. A later edition of this Code may be used if a written reconciliation is performed.	ACI-349-97 recognizes increased structural role of the ISFSI pad in an anchored cask storage configuration and imposes requirements on embedment design.
3. Limitations on the pad design parameters	Per Table 2.2.9	Per Table 2.0.4	In free-standing cask storage, the non-mechanistic tipover requirement limits the stiffness of the pad. In the anchored storage configuration, increased pad stiffness is permitted; however, the permissible HI-STORM carry height is reduced.
4. HI-STORM Carry Height	11 inches (for ISFSI pad parameter Set A or Set B) or, otherwise, site-specific. Not applicable if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.	Determined site-specifically. Not applicable if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.	Appendix 3.A provides the technical basis for free-standing installation. Depending on the final ISFSI pad configuration (thickness, concrete strength, subgrade, etc.), and the method of transport, an allowable carry height may need to be established.

TABLE 2.0.5 (continued)  
ISFSI PAD REQUIREMENTS FOR FREE-STANDING AND ANCHORED HI-STORM INSTALLATION

<i>Item</i>	<i>Free-Standing</i>	<i>Anchored</i>	<i>Comments</i>
5. Maximum seismic input on the pad/cask contact surface. $G_H$ is the vectorial sum of the two horizontal ZPAs and $G_V$ is the vertical ZPA	$G_H + \mu G_V \leq \mu$	$G_H \leq 2.12$  AND  $G_V \leq 1.5$	
6. Required minimum value of cask to pad static coefficient of friction ( $\mu$ , must be confirmed by testing).	Greater than or equal to 0.53 (per Table 2.2.9).	Same as that for free-standing condition	
7. Applicable Wind and Large Missile Loads	Per Table 2.2.4, missile and wind loading different from the tabulated values, require 10CFR 72.48 evaluation	The maximum overturning moment at the base of the cask due to lateral missile and/or wind action must be less than $1 \times 10^7$ ft-lb.	The bases are provided in Section 3.4.8 for free-standing casks; the limit for anchored casks ensures that the anchorage system will have the same structural margins established for seismic loading.
8. Small and medium missiles (penetrant missile)	Per Table 2.2.5, missiles and wind loading different from the tabulated value, require 10CFR 72.48 evaluation.	Same as for free-standing cask construction.	
9. Design Loadings for the ISFSI Pad	Per load combinations in Section 2.0.4 using site-specific load.	Same as for free-standing cask.	

## 2.1 SPENT FUEL TO BE STORED

### 2.1.1 Determination of The Design Basis Fuel

The HI-STORM 100 System is designed to store most types of fuel assemblies generated in the commercial U.S. nuclear industry. Boiling-water reactor (BWR) fuel assemblies have been supplied by The General Electric Company (GE), Siemens, Exxon Nuclear, ANF, UNC, ABB Combustion Engineering, and Gulf Atomic. Pressurized-water reactor (PWR) fuel assemblies are generally supplied by Westinghouse, Babcock & Wilcox, ANF, and ABB Combustion Engineering. ANF, Exxon, and Siemens are historically the same manufacturing company under different ownership. Within this report, SPC is used to designate fuel manufactured by ANF, Exxon, or Siemens. Publications such as Refs. [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors. A central object in the design of the HI-STORM 100 System is to ensure that a majority of SNF discharged from the U.S. reactors can be stored in one of the MPCs.

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

- i. The South Texas Units 1 & 2 SNF, and CE 16x16 System 80 SNF are too long to be accommodated in the available MPC cavity length.

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

In summary, the geometric compatibility of the SNF with the MPC designs does not require the definition of a design basis fuel assembly. This, however, is not the case for structural, confinement, shielding, thermal-hydraulic, and criticality criteria. In fact, a particular fuel type in a category (PWR or BWR) may not control the cask design in all of the above-mentioned criteria. To ensure that no SNF listed in Refs. [2.1.1] and [2.1.2] which is geometrically admissible in the MPC is precluded, it is necessary to determine the governing fuel specification for each analysis criterion. To make the necessary determinations, potential candidate fuel assemblies for each qualification criterion were considered. Table 2.1.1 lists the PWR fuel assemblies that were evaluated. These fuel assemblies were evaluated to define the governing design criteria for PWR fuel. The BWR fuel assembly designs evaluated are listed in Table 2.1.2. Tables 2.1.3 and 2.1.4 provide the fuel characteristics 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. ~~These 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 ~~decay heat generation~~ *thermal*. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Additional information on the design basis fuel definition is presented in the following subsections.

### 2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STORM 100 System is intact ~~Zircaloy-ZR or stainless steel (SS) clad fuel assemblies with the characteristics listed in Tables 2.1.6 through 2.1.24. or intact stainless steel clad fuel assemblies with the characteristics listed in Table 2.1.8.~~ The placement of a single stainless steel clad fuel assembly in a MPC necessitates that all fuel assemblies (stainless steel clad or Zircaloy-ZR clad) stored in that MPC meet the maximum heat generation requirements for stainless steel clad fuel specified in Table 2.1.8. ~~Intact BWR MOX fuel assemblies shall meet the requirements of Table 2.1.7.~~

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 ~~the Approved Contents section of Appendix B to the CoC~~ *Section 2.1.9* can be safely stored in the HI-STORM 100 System.

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.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with 250 x 250 fine mesh screens, for storage in the HI-STORM 100 System (*see Figures 2.1.1 and 2.1.2B, C, and D*). The MPC-24E and MPC 32 ~~is~~ *are* designed to accommodate PWR damaged fuel. The MPC-24EF and MPC-32F ~~is~~ *are* designed to accommodate PWR damaged fuel and fuel debris. The MPC-68 is designed to accommodate BWR damaged fuel. The 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 ~~authorized for loading in the HI-STORM 100 System are provided in Table~~

~~2.1.7 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*, ~~the Approved Contents section of Appendix B to the CoC~~. 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).

*MPC contents classified as fuel debris are required to be stored in DFCs and in the applicable “F” model MPC as specified in Section 2.1.9. The “F” (or “FF”) indicates the MPC is qualified for storage of intact fuel, damaged fuel, and fuel debris, in quantities and locations specified in Section 2.1.9. The basket designs for the standard and “F” model MPCs are identical. The lid and shell designs of the “F” models are unique in that the upper shell portion of the canister is thickened for additional strength needed under hypothetical accident conditions of transportation under 10 CFR 71. This design feature is not required for dry storage, but must be considered in fuel loading for dry storage to ensure the dual purpose function of the MPC by eliminating the need to re-package the fuel for transportation. Figure 2.1.9 shows the details of the differences between the standard and “F” model MPC shells. These details are common for both the PWR and BWR series MPC models.*

#### 2.1.4 Deleted

#### 2.1.5 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are ~~listed in Tables 2.1.6, 2.1.7, and 2.1.8~~ *specified in Section 2.1.9*. The centers of gravity reported in Section 3.2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10. In order to qualify for storage in the MPC, the SNF must satisfy the physical parameters listed in ~~Tables 2.1.6, 2.1.7, or 2.1.8~~ *Section 2.1.9*.

#### 2.1.6 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly, ~~the allowable fuel cladding temperature based on cooling time,~~ and the decay heat removal capabilities of the HI-STORM 100 System. *No attempt is made to link the maximum allowable decay heat per fuel assembly with burnup, enrichment, or cooling time. Rather, the decay heat per fuel assembly is adjusted to yield peak fuel cladding temperatures with an allowance for margin to the temperature limit. The same fuel assembly decay heats are used for all fuel assembly designs within a given class of fuel assemblies (i.e., ZR clad PWR, stainless steel clad BWR, etc.).* ~~The maximum heat generation rate per assembly for the design basis fuel assembly is based on the fuel assembly type with the highest decay heat for a given enrichment, burnup, and cooling time. This decay heat design basis fuel~~

assembly is listed in Table 2.1.5. Section 5.2 describes the method used to determine the design basis fuel assembly type and calculate the decay heat load.

To ensure the permissible fuel cladding temperature limits are not exceeded, the Approved Contents section of Appendix B to the CoC Section 2.1.9 specifies the allowable decay heat per assembly for each MPC model. For both uniform and regionalized loading of moderate and high burnup Zircaloy clad fuel assemblies, the allowable decay heat per assembly is a function of cooling time and is presented in Appendix B to the CoC Section 2.1.9 in Tables 2.1.5 and 2.1.7. For stainless steel clad fuel assemblies, the allowable decay heat per assembly is not dependent upon cooling time and is specified in Table 2.1.1 of Appendix B to the CoC. Due to the large conservatisms in the thermal evaluations and the relatively long cooling times and corresponding low decay heats for stainless steel clad fuel, an age dependent allowable decay heat limit is not necessary.

The specified decay heat load can be attained by varying burnups and cooling times. The Approved Contents section of Appendix B to the CoC provides the burnup and cooling time limits for intact Zircaloy/ZR clad fuel to meet the thermal requirements for the various MPC's.

The Approved Contents section of Appendix B to the CoC Section 2.1.9 also includes separate cooling time, burnup, and decay heat limits for uniform fuel loading and regionalized fuel loading. Regionalized loading allows higher heat emitting fuel assemblies to be stored in the center fuel storage locations than would otherwise be authorized for storage under uniform loading conditions.

The fuel cladding temperature is also affected by *the heat transfer characteristics of the fuel assemblies. There is no single fuel assembly design used in all thermal calculations that is bounding of all others. Instead, each thermal calculation, comprising the overall thermal analysis presented in Chapter 4, was performed using the fuel assembly design that results in the most conservative result for the individual calculation. By always using the fuel assembly design that is most conservative for a particular calculation, it is ensured that each calculation is bounding for all fuel assembly designs. The bounding fuel assembly design for each thermal calculation and fuel type is provided in Table 2.1.5.* other factors. A governing geometry that maximizes the impedance to the transmission of heat out of the fuel rods has been defined. The governing thermal parameters to ensure that the range of SNF discussed previously are bounded by the thermal analysis are discussed in detail and specified in Chapter 4. By utilizing these bounding thermal parameters, the calculated peak fuel rod cladding temperatures are conservative for actual spent fuel assemblies which have greater thermal conductivities.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in Refs. [2.1.7] and [2.1.8] are utilized and summarized in Table 2.1.11 and Figures 2.1.3 and 2.1.4 for reference. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

*Except for MPC-68F,* fuel may be stored in the MPC using one of two storage strategies, namely, uniform loading and regionalized loading. Uniform loading allows storage of any fuel assembly in any fuel storage location, subject to additional restrictions, *such as those specified in the CoC* for

~~preferential fuel loading and~~ loading of fuel assemblies containing non-fuel hardware as defined in Table 1.0.1. Regionalized fuel loading allows for higher heat emitting fuel assemblies to be stored in the central core basket storage locations (*inner region*) with lower heat emitting fuel assemblies in the peripheral fuel storage locations (*outer region*). Regionalized loading allows storage of higher heat emitting fuel assemblies than would otherwise be permitted using the uniform loading strategy. The definition of the regions for each MPC model ~~and the associated burnup, cooling time, and decay heat limits are found~~ provided in ~~Appendix B to the CoC~~Table 2.1.13. Regionalized fuel loading ~~meets the intent of preferential fuel loading.~~ *is not permitted in MPC-68F.*

#### 2.1.7 Radiological Parameters for Design Basis SNF

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

The gamma and neutron sources are separate and are affected differently by enrichment, burnup, and cooling time. It is recognized that, at a given burnup, the radiological source terms increase monotonically as the initial enrichment is reduced. The shielding design basis fuel assembly, therefore, is evaluated at conservatively high burnups, low cooling times, and low enrichments, as discussed in Chapter 5. The shielding design basis fuel assembly thus bounds all other fuel assemblies.

The design basis dose rates can be met by a variety of burnup levels and cooling times. ~~The Approved Contents section of Appendix B to the CoC~~Section 2.1.9 provides the *procedure for determining* burnup and cooling time limits for all of the authorized fuel assembly array/classes for both uniform fuel loading and regionalized loading.

Table 2.1.11 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Thoria Rod Canister is authorized for storage in combination with other intact and damaged fuel, and fuel debris as specified in ~~Appendix B to the CoC~~Section 2.1.9.

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

#### 2.1.8 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68, MPC-68F, MPC-68FF, ~~and~~ 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  $^{10}\text{B}$  areal density in the neutron absorber panels for each MPC model is shown in Table 2.1.15.

For all MPCs, the  $^{10}\text{B}$  areal density used for the criticality analysis is conservatively established below at 75% of 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.  $^{10}\text{B}$  areal density to demonstrate that the reactivity under the most adverse accumulation of tolerances and biases is less than 0.95. This is consistent with NUREG-1536 [2.1.5] which requires-suggests a 25% reduction in  $^{10}\text{B}$  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. A large body of sampling data accumulated by Holtec from thousands of manufactured Boral panels indicates the average  $^{10}\text{B}$  areal densities to be approximately 15% greater than the specified minimum.

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 provides the required soluble boron concentrations for these MPCs. Minimum soluble boron concentration is also included as Limiting Condition for Operation (LCO) 3.3.1 in the Technical Specifications found in Appendix A to the CoC.

#### 2.1.9 Summary of SNF Design Criteria Authorized Contents

Tables 2.1.1 through 2.1.8 and Table 2.1.12 provide the design characteristics Tables 2.1.3, 2.1.4, 2.1.12, and 2.1.17 through 2.1.29 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. Much of this information is repeated in the Approved Contents section of Appendix B to the CoC. Only fuel meeting the specifications in the CoC is authorized for storage. 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. Limits on decay heat, burnup, and cooling time for stainless steel-clad fuel are provided in Tables 2.1.17 through 2.1.24.

##### 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<sup>†</sup>. 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 12.2.10 describes how compliance with these limits may be verified, including practical examples.

#### 2.1.9.1.1 Uniform Fuel Loading Decay Heat Limits for ZR-Clad Fuel

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.

#### 2.1.9.1.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.1.27 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in both the inner and outer regions for regionalized fuel loading in each MPC model.

#### 2.1.9.1.3 Burnup 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

<sup>†</sup> Note that the stainless steel-clad fuel 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 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.

- Choose a fuel assembly minimum enrichment,  $E_{235}$ .
- 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 2.1.9.3

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 or 2.1.9.2 (kW)

$E_{235}$  = Minimum fuel assembly average enrichment (wt. %  $^{235}\text{U}$ )  
(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 Other 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.*

Table 2.1.1

PWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

<b>Assembly Class</b>	<b>Array Type</b>
B&W 15x15	All
B&W 17x17	All
CE 14x14	All
CE 16x16	All except System 80™
WE 14x14	All
WE 15x15	All
WE 17x17	All
St. Lucie	All
Ft. Calhoun	All
Haddam Neck (Stainless Steel Clad)	All
San Onofre 1 (Stainless Steel Clad)	All
Indian Point 1	All

Table 2.1.2

BWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	Array Type			
GE BWR/2-3	All 7x7	All 8x8	All 9x9	All 10x10
GE BWR/4-6	All 7x7	All 8x8	All 9x9	All 10x10
Humboldt Bay	All 6x6	All 7x7 (Zircaloy-ZR Clad)		
Dresden-1	All 6x6	All 8x8		
LaCrosse (Stainless Steel Clad)	All			

Table 2.1.3  
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	14x14 A	14x14 B	14x14 C	14x14 D	14x14E
Clad Material (Note 2)	<del>Zr</del> ZR	ZR <del>Zr</del>	ZR <del>Zr</del>	SS	SS
Design Initial U (kg/assy.) (Note 3)	$\leq 407\ 365$	$\leq 407\ 412$	$\leq 425\ 438$	$\leq 400$	$\leq 206$
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.6$ (24) $\leq 5.0$ (24E/24EF)	$\leq 4.0$ (24) $\leq 5.0$ (24E/24EF)	$\leq 5.0$ (24) $\leq 5.0$ (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % <sup>235</sup> U)	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$	$\leq 5.0$
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	$\geq 0.400$	$\geq 0.417$	$\geq 0.440$	$\geq 0.422$	$\geq 0.3415$
Fuel Clad I.D. (in.)	$\leq 0.3514$	$\leq 0.3734$	$\leq 0.3880$	$\leq 0.3890$	$\leq 0.3175$
Fuel Pellet Dia. (in.)	$\leq 0.3444$	$\leq 0.3659$	$\leq 0.3805$	$\leq 0.3835$	$\leq 0.3130$
Fuel Rod Pitch (in.)	$\leq 0.556$	$\leq 0.556$	$\leq 0.580$	$\leq 0.556$	Note 6
Active Fuel Length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 144$	$\leq 102$
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	$\geq 0.017$	$\geq 0.017$	$\geq 0.038$	$\geq 0.0145$	N/A

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

Fuel Assembly Array/Class	15x15 A	15x15 B	15x15 C	15x15 D	15x15 E	15x15 F
Clad Material (Note 2)	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>
Design Initial U (kg/assy.) (Note 3)	≤ 464 473	≤ 464 473	≤ 464 473	≤ 475 495	≤ 475 495	≤ 475 495
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

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

Fuel Assembly Array and Class	15x15 G	15x15H	16x16 A	17x17A	17x17 B	17x17 C
Clad Material (Note 2)	SS	ZRZ <del>±</del>	ZRZ <del>±</del>	ZRZ <del>±</del>	ZRZ <del>±</del>	ZRZ <del>±</del>
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ <del>475</del> 495	≤ <del>443</del> 448	≤ <del>467</del> 433	≤ <del>467</del> 474	≤ <del>474</del> 480
Initial Enrichment (MPC-24, 24E, and 24EF without soluble boron credit) (wt % <sup>235</sup> U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit — see Notes 5 and 7) (wt % <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≥ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020



Table 2.1.3 (continued)  
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. ~~Zr designates cladding material made of zirconium or zirconium alloys~~ See Table 1.0.1 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 PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances.
4. Each guide tube replaces four fuel rods.
5. Soluble boron concentration per ~~Technical Specification LCO 3.3.1~~ Tables 2.1.14 and 2.1.16, as applicable.
6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.
7. For those MPCs loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum initial enrichment of the intact fuel assemblies, damaged fuel assemblies and fuel debris is 4.0 wt.% <sup>235</sup>U.

Table 2.1.4  
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	6x6 A	6x6 B	6x6 C	7x7 A	7x7 B	8x8 A
Clad Material (Note 2)	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>	ZRZ <del>†</del>
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ <del>195</del> 198	≤ 120
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 2.7	≤ 2.7 for UO <sub>2</sub> rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

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

<b>Fuel Assembly Array and Class</b>	<b>8x8 B</b>	<b>8x8 C</b>	<b>8x8 D</b>	<b>8x8 E</b>	<b>8x8F</b>	<b>9x9 A</b>
Clad Material (Note 2)	ZRZ <del>F</del>	ZRZ <del>F</del>	ZRZ <del>F</del>	ZRZ <del>F</del>	ZRZ <del>F</del>	ZRZ <del>F</del>
Design Initial U (kg/assy.) (Note 3)	$\leq$ <del>191</del> 192	$\leq$ <del>191</del> 190	$\leq$ <del>191</del> 190	$\leq$ <del>191</del> 190	$\leq$ 191	$\leq$ <del>179</del> 180
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.0	$\leq$ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	$\geq$ 0.4840	$\geq$ 0.4830	$\geq$ 0.4830	$\geq$ 0.4930	$\geq$ 0.4576	$\geq$ 0.4400
Fuel Clad I.D. (in.)	$\leq$ 0.4295	$\leq$ 0.4250	$\leq$ 0.4230	$\leq$ 0.4250	$\leq$ 0.3996	$\leq$ 0.3840
Fuel Pellet Dia. (in.)	$\leq$ 0.4195	$\leq$ 0.4160	$\leq$ 0.4140	$\leq$ 0.4160	$\leq$ 0.3913	$\leq$ 0.3760
Fuel Rod Pitch (in.)	$\leq$ 0.642	$\leq$ 0.641	$\leq$ 0.640	$\leq$ 0.640	$\leq$ 0.609	$\leq$ 0.566
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	$\geq$ 0.034	$>$ 0.00	$>$ 0.00	$\geq$ 0.034	$\geq$ 0.0315	$>$ 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.100	$\leq$ 0.100	$\leq$ 0.120

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

Fuel Assembly Array and Class	9x9 B	9x9 C	9x9 D	9x9 E (Note 13)	9x9 F (Note 13)	9x9 G
Clad Material (Note 2)	ZRZ†	ZRZ†	ZRZ†	ZRZ†	ZRZ†	ZRZ†
Design Initial U (kg/assy.) (Note 3)	≤ 479 180	≤ 479 182	≤ 479 182	≤ 479 183	≤ 479 183	≤ 479 164
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

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

<b>Fuel Assembly Array and Class</b>	<b>10x10 A</b>	<b>10x10 B</b>	<b>10x10 C</b>	<b>10x10 D</b>	<b>10x10 E</b>
Clad Material (Note 2)	ZRZ <del>z</del>	ZRZ <del>z</del>	ZRZ <del>z</del>	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ <del>188</del> 179	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% <sup>235</sup> U)	≤ 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
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1.4 (continued)  
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.
2. *See Table 1.0.1 for the definition of "ZR."* ~~Zr designates cladding material made of zirconium or zirconium alloys.~~
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.  $\leq 0.635$  wt. %  $^{235}\text{U}$  and  $\leq 1.578$  wt. % total fissile plutonium ( $^{239}\text{Pu}$  and  $^{241}\text{Pu}$ ), (wt. % of total fuel weight, i.e.,  $\text{UO}_2$  plus  $\text{PuO}_2$ )
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 or clad O.D., clad I.D., and pellet diameter.
14. For those MPCs 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.%  $^{235}\text{U}$ , as applicable.

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

<b>Criterion</b>	<b>MPC-68/68F/68FFBWR</b>	<b>MPC-24PWR</b>
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Array/Class 10x10A)	B&W 15x15 (Array/Class 15x15F)
Source Term (Shielding)	GE 7x7 (Class 7x7B)	B&W 15x15 (Class 15x15F)
<i>Fuel Assembly Effective Planar Thermal Conductivity</i>	<i>GE-11 9x9</i>	<i><u>W</u> 17x17 OFA</i>
<i>Fuel Basket Effective Axial Thermal Conductivity</i>	<i>GE 7x7</i>	<i><u>W</u> 14x14 OFA</i>
<i>MPC Density and Heat Capacity</i>	<i>Dresden 6x6</i>	<i><u>W</u> 14x14 OFA</i>
<i>MPC Fuel Basket Axial Resistance to Thermosiphon Flow</i>	<i>GE-11 9x9</i>	<i><u>W</u> 17x17 OFA</i>

Table 2.1.6

TABLE INTENTIONALLY DELETED

DESIGN CHARACTERISTICS FOR ~~INTACT ZIRCALOY ZR CLAD~~  
FUEL ASSEMBLIES<sup>1</sup>

	<del>MPC 68/68FF</del>	<del>MPC 68F</del>	<del>MPC 24</del>	<del>MPC 24E/24EF</del>	<del>MPC 32/32F</del>
<b>PHYSICAL PARAMETERS:</b>					
Max. assembly width (in.)	5.85	4.70	8.54	8.54	8.54
Max. assembly length (in.)	176.5	135.0	176.8	176.8	176.8
Max. assembly weight <sup>2</sup> (lb.)	700	400	1680	1680	1680
Max. active fuel length (in.)	150	120	150	150	150
<b>RADIOLOGICAL AND THERMAL CHARACTERISTICS:</b>					
	<del>MPC 68/68FF</del>	<del>MPC 68F</del>	<del>MPC 24</del>	<del>MPC 24E/24EF</del>	<del>MPC 32/32F</del>
Max. initial enrichment (wt% <sup>235</sup> U)	See Table 2.1.4	See Table 2.1.4	See Table 2.1.3	See Table 2.1.3	See Table 2.1.3
Max. heat generation (W)	Table 2.0.1	Table 2.0.1	Table 2.0.1	Table 2.0.1	Table 2.0.1
Max. average burnup (MWD/MTU) <sup>3</sup>	59,900-70,000	30,000	66,200-75,000	68,200-75,000	54,700-75,000
Min. cooling time (years)	5-3	18	5-3	5-3	5-3

<sup>1</sup> — These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.

<sup>2</sup> — Fuel assembly weight including non-fuel hardware, and channels, as applicable, based on DOE MPC DPS [2.1.6].

<sup>3</sup> — The maximum burnup for fuel assemblies with cladding made of materials other than Zircaloy 2 or Zircaloy 4 is 45,000 MWD/MTU



Table 2.1.7

TABLE INTENTIONALLY DELETED

**DESIGN CHARACTERISTICS FOR DAMAGED  
FUEL ASSEMBLIES<sup>1</sup>**

	MPC-68/68FF (Damaged Fuel and Fuel Debris)	MPC-68F (Damaged Fuel and Fuel Debris)	MPC-24E/24EF MPC-32/MPC-32F (Damaged Fuel and Fuel Debris)
<b>PHYSICAL PARAMETERS:</b>			
Max. assembly width (in.)	5.5	4.7	8.54
Max. assembly length (in.)	176.5	135.0	176.8
Max. assembly weight <sup>2</sup> (lb.)	700	550	1680
Max. active fuel length (in.)	150	110	150
Fuel rod clad material	Zircaloy ZR/SS	Zircaloy ZR	Zircaloy ZR/SS
<b>RADIOLOGICAL AND THERMAL CHARACTERISTICS:</b>			
Max. heat generation (W)	356 See Table 2.0.1	115	927 See Table 2.0.1
Min. cooling time (yr)	5-3	18	5-3
Max. initial enrichment (wt.% <sup>235</sup> U) for UO <sub>2</sub> rods	4.0	2.7	4.0-5.0
Max. initial enrichment for MOX rods	0.635 wt.% <sup>235</sup> U 1.578 wt.% Total Fissile Plutonium	0.635 wt.% <sup>235</sup> U 1.578 wt.% Total Fissile Plutonium	N/A
Max. average burnup (MWD/MTU) <sup>3</sup>	59,90070,000 (ZR) 22,500 (SS)	30,000	68,20075,000 (ZR) 40,000 (SS)

Note: Refer to the Approved Contents section of Appendix B to the CoC for restrictions on the number and location of damaged fuel assemblies and fuel debris authorized for loading in the HI-STORM 100 System.

<sup>1</sup> These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.

<sup>2</sup> Fuel assembly weight including non-fuel hardware, channels, and DFC, as applicable, based on DOE MPC DPS [2.1.6].

<sup>3</sup> The maximum burnup for fuel assemblies with cladding made of materials other than Zircaloy 2 or Zircaloy 4 is 45,000 MWD/MTU

Table 2.1.8

TABLE INTENTIONALLY DELETED

DESIGN CHARACTERISTICS FOR INTACT STAINLESS STEEL CLAD FUEL ASSEMBLIES<sup>†</sup>

	BWR MPC 68/68FF	PWR MPC 24/24E/24EF	PWR MPC 32/32F
<b>PHYSICAL PARAMETERS:</b>			
Max. assembly width <sup>2</sup> (in.)	5.62	8.54	8.54
Max. assembly length <sup>2</sup> (in.)	102.5	176.8	176.8
Max. assembly weight <sup>3</sup> (lb.)	700	1680	1680
Max. active fuel length <sup>2</sup> (in.)	83	144	144
<b>RADIOLOGICAL AND THERMAL CHARACTERISTICS :</b>			
Max. heat generation (W)	95	710	500
Min. cooling time (yr)	10	8	9/20
Max. initial enrichment without soluble boron credit (wt.% <sup>235</sup> U)	4.0	See Table 2.1.3	N/A
Max. initial enrichment with soluble boron credit (wt.% <sup>235</sup> U)	N/A	5.0	5.0
Max. average burnup (MWD/MTU)	22,500	40,000	30,000/40,000

<sup>†</sup> — These are limiting values for all authorized fuel assembly array/classes. Refer to the Approved Contents section of Appendix B to the CoC for specific limits for each fuel assembly array/class.

<sup>2</sup> — Unirradiated nominal dimensions are shown.

<sup>3</sup> — Fuel assembly weight including non fuel hardware and channels, as applicable, based on DOE MPC DPS [2.1.6].

Table 2.1.9

## SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length w/o NFH <sup>1</sup> (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
CE 14x14	157	4.1	137	9.5	10.0
CE 16x16	176.8	4.7	150	0	0
BW 15x15	165.7	8.4	141.8	6.7	4.1
W 17x17 OFA	159.8	3.7	144	8.2	8.5
W 17x17 Std	159.8	3.7	144	8.2	8.5
W 17x17 V5H	160.1	3.7	144	7.9	8.5
W 15x15	159.8	3.7	144	8.2	8.5
W 14x14 Std	159.8	3.7	145.2	9.2	7.5
W 14x14 OFA	159.8	3.7	144	8.2	8.5
Ft. Calhoun	146	6.6	128	10.25	20.25
St. Lucie 2	158.2	5.2	136.7	10.25	8.05
B&W 15x15 SS	137.1	3.873	120.5	19.25	19.25
W 15x15 SS	137.1	3.7	122	19.25	19.25
W 14x14 SS	137.1	3.7	120	19.25	19.25
Indian Point 1	137.2	17.705	101.5	18.75	20.0

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

<sup>1</sup> NFH is an abbreviation for non-fuel hardware, including control components. Fuel assemblies with control components may require shorter fuel spacers.

Table 2.1.10

## SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
GE/2-3	171.2	7.3	150	4.8	0
GE/4-6	176.2	7.3	150	0	0
Dresden 1	134.4	11.2	110	18.0	28.0
Humboldt Bay	95.0	8.0	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	142.1 <sup>†</sup>	11.2	110	17.0	16.9
Humboldt Bay Damaged Fuel or Fuel Debris	105.5 <sup>†</sup>	8.0	79	35.25	35.25
LaCrosse	102.5	10.5	83	37.0	37.5

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

<sup>†</sup> Fuel assembly length includes the damaged fuel container.

Table 2.1.11  
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

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

<sup>1</sup> Reference 2.1.7  
<sup>2</sup> Reference 2.1.8

Table 2.1.12

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

PARAMETER	MPC-68 or MPC-68F
Cladding Type	Zircaloy-(Zr)
Composition	98.2 wt.% ThO <sub>2</sub> , 1.8 wt.% UO <sub>2</sub> with an enrichment of 93.5 wt. % <sup>235</sup> U
Number of Rods Per Thoria Canister	≤ 18
Decay Heat Per Thoria Canister	≤ 115 Watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time ≥ 18 years and average burnup ≤ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	≤ 27 kg/canister
Fuel Cladding O.D.	≥ 0.412 inches
Fuel Cladding I.D.	≤ 0.362 inches
Fuel Pellet O.D.	≤ 0.358 inches
Active Fuel Length	≤ 111 inches
Canister Weight	≤ 550 lbs., including Thoria Rods
<i>Canister Material</i>	<i>Type 304 SS</i>

Table 2.1.13  
MPC Fuel Loading Regions

<b>MPC MODEL</b>	<b>REGION 1 FUEL STORAGE LOCATIONS*</b>	<b>REGION 2 FUEL STORAGE LOCATIONS</b>
MPC-24, 24E and 24EF	9, 10, 15, and 16	All Other Locations
MPC-32/32F	7, 8, 12 through 15, 18 through 21, 25, and 26	All Other Locations
MPC-68/68F/68FF	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All Other Locations

\*Note: Refer to Figures 1.2.2 through 1.2.4

Table 2.1.14

Soluble Boron Requirements for PWR MPC-24/24E/24EF Fuel Wet Loading and Unloading Operations

MPC MODEL	FUEL ASSEMBLY MAXIMUM AVERAGE ENRICHMENT (wt % <sup>235</sup> U)	MINIMUM SOLUBLE BORON CONCENTRATION (ppmb)
MPC-24	All fuel assemblies with initial enrichment <sup>†</sup> less than the prescribed value for soluble boron credit	0
MPC-24	One or more fuel assemblies with an initial enrichment <sup>†</sup> greater than or equal to the prescribed value for no soluble boron credit AND $\leq 5.0$ wt. %	$\geq 400$
MPC-24E/24EF	All fuel assemblies with initial enrichment <sup>†</sup> less than the prescribed value for soluble boron credit	0
MPC-24E/24EF	<i>All fuel assemblies classified as intact fuel assemblies and <math>\Theta</math> one or more fuel assemblies with an initial enrichment<sup>†</sup> greater than or equal to the prescribed value for no soluble boron credit AND <math>\leq 5.0</math> wt. %</i>	$\geq 300$
<i>MPC-24E/24EF</i>	<i>One or more fuel assemblies classified as damaged fuel or fuel debris and one or more fuel assemblies with initial enrichment <math>&gt; 4.0</math> wt.% and <math>\leq 5.0</math> wt.%</i>	$\geq 600$
MPC-32	All fuel assemblies with initial enrichment $\leq 4.1$ wt. %	$\geq 1900$
MPC-32	One or more fuel assemblies with an initial enrichment $> 4.1$ and $\leq 5.0$ wt. %	$\geq 2600$

<sup>†</sup> See Table 2.1.3 for these enrichments.



Table 2.1.15

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

MPC MODEL	MINIMUM <del>B-10</del> $^{10}\text{B}$ LOADING (g/cm <sup>2</sup> )	
	<i>Boral Neutron Absorber Panels</i>	<i>METAMIC Neutron Absorber Panels</i>
MPC-24	0.0267	0.0223
MPC-24E and MPC-24EF	0.0372	0.0310
MPC-32/32F	0.0372	0.0310
MPC-68 and MPC-68FF	0.0372	0.0310
MPC-68F	0.01	N/A (Note 1)

Notes:

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

Table 2.1.16

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

<i>Fuel Assembly Array/Class</i>	<i>All Intact Fuel Assemblies</i>		<i>One or More Damaged Fuel Assemblies or Fuel Debris</i>	
	<i>Initial Enrichment <math>\leq 4.1</math> wt. % <math>^{235}\text{U}</math> (ppmb)</i>	<i>Initial Enrichment <math>&gt; 4.1</math> wt. % and <math>\leq 5.0</math> wt. % <math>^{235}\text{U}</math> (ppmb)</i>	<i>Initial Enrichment <math>\leq 4.1</math> wt. % <math>^{235}\text{U}</math> (ppmb)</i>	<i>Initial Enrichment <math>&gt; 4.1</math> wt. % and <math>\leq 5.0</math> wt. % <math>^{235}\text{U}</math> (ppmb)</i>
<i>14x14A/B/C/D/E</i>	<i>1,300</i>	<i>1,900</i>	<i>1,500</i>	<i>2,300</i>
<i>15x15A/B/C/G</i>	<i>1,800</i>	<i>2,500</i>	<i>1,900</i>	<i>2,700</i>
<i>15x15D/E/F/H</i>	<i>1,900</i>	<i>2,600</i>	<i>2,100</i>	<i>2,900</i>
<i>16x16A</i>	<i>1,300</i>	<i>1,900</i>	<i>1,500</i>	<i>2,300</i>
<i>17x17A/B/C</i>	<i>1,900</i>	<i>2,600</i>	<i>2,100</i>	<i>2,900</i>

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

<b>PARAMETER</b>	<b>VALUE</b>
<i>Fuel Type</i>	<i>Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class</i>
<i>Maximum Initial Enrichment per Assembly</i>	<i>As specified in Table 2.1.3 for the applicable array/class</i>
<i>Post-irradiation Cooling Time and Average Burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1 SS clad: <math>\geq 8</math> years and <math>\leq 40,000</math> MWD/MTU</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1 SS clad: <math>\leq 710</math> Watts</i>
<i>Non-Fuel Hardware Burnup and Cooling Time</i>	<i>As specified in Table 2.1.25</i>
<i>Fuel Assembly Length</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq 1,680</math> lbs (including non-fuel hardware)</i>
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity is limited to up to 24 PWR intact fuel assemblies.</i></li> <li>▪ <i>Neutron sources, damaged fuel assemblies and fuel debris are not permitted for storage in MPC-24.</i></li> <li>▪ <i>BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</i></li> <li>▪ <i>CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16</i></li> <li>▪ <i>Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.</i></li> </ul>

Table 2.1.18

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>			
<i>Fuel Type(s)</i>	<i>Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels</i>	<i>Uranium oxide, BWR damaged fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, placed in Damaged Fuel Containers (DFCs)</i>	<i>Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without channels</i>	<i>Mixed Oxide (MOX) BWR damaged fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without channels, placed in Damaged Fuel Containers (DFCs)</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class</i>	<i>ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class</i>	<i>ZR</i>	<i>ZR</i>
<i>Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment</i>	<i>As specified in Table 2.1.4 for the applicable array/class</i>	<i>Planar Average:  <math>\leq 2.7 \text{ wt}\% \text{ }^{235}\text{U}</math>  <i>for array/classes 6x6A, 6x6C, 7x7A, and 8x8A;</i>   <math>\leq 4.0 \text{ wt}\% \text{ }^{235}\text{U}</math>  <i>for all other array/classes</i>   <i>Rod:</i>   <i>As specified in Table 2.1.4</i></i>	<i>As specified in Table 2.1.4 for array/class 6x6B</i>	<i>As specified in Table 2.1.4 for array/class 6x6B</i>

Table 2.1.18 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>			
<i>Post-irradiation Cooling Time and Average Burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: Note 4</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: Note 4.</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTIHM.</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTIHM.</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: <math>\leq</math> 95 Watts</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: <math>\leq</math> 95 Watts</i>	$\leq$ 115 Watts	$\leq$ 115 Watts
<i>Fuel Assembly Length</i>	$\leq$ 176.5 in. (nominal design)	<i>Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: <math>\leq</math> 135.0 in. (nominal design)</i>  <i>All Other array/classes: <math>\leq</math> 176.5 in. (nominal design)</i>	$\leq$ 135.0 in. (nominal design)	$\leq$ 135.0 in. (nominal design)
<i>Fuel Assembly Width</i>	$\leq$ 5.85 in. (nominal design)	<i>Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: <math>\leq</math> 4.7 in. (nominal design)</i>  <i>All Other array/classes: <math>\leq</math> 5.85 in. (nominal design)</i>	$\leq$ 4.70 in. (nominal design)	$\leq$ 4.70 in. (nominal design)

Table 2.1.18 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	VALUE (Note 1)			
Fuel Assembly Weight	$\leq 700$ lbs. (including channels)	Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: $\leq 550$ lbs. (including channels and DFC)	$\leq 400$ lbs, including channels	$\leq 550$ lbs, including channels and DFC
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12 plus any combination of array/class 6x6A, 6x6B, 6x6C, 7x7A, and/or 8x8A damaged fuel assemblies in DFCs and intact fuel assemblies up to a total of 68.</li> <li>▪ Up to 16 damaged fuel assemblies from plants other than Dresden Unit 1 or Humboldt Bay may be stored in DFCs 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 up to a total of 68</li> <li>▪ 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.</li> <li>▪ 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.</li> <li>▪ Fuel debris is not permitted for storage in MPC-68.</li> </ul>			

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, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time  $\geq 18$  years, an average burnup  $\leq 30,000$  MWD/MTU, and a maximum decay heat  $\leq 115$  Watts.
3. Array/class 8x8F fuel assemblies shall have a cooling time  $\geq 10$  years, an average burnup  $\leq 27,500$  MWD/MTU, and a maximum decay heat  $\leq 183.5$  Watts.
4. SS-clad fuel assemblies shall have a cooling time  $\geq 10$  years, and an average burnup  $\leq 22,500$  MWD/MTU.

Table 2.1.19

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

<b>PARAMETER</b>	<b>VALUE (Notes 1 and 2)</b>			
<i>Fuel Type(s)</i>	<i>Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels</i>	<i>Uranium oxide, BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)</i>	<i>Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels</i>	<i>Mixed Oxide (MOX) BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)</i>
<i>Cladding Type</i>	<i>ZR</i>	<i>ZR</i>	<i>ZR</i>	<i>ZR</i>
<i>Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment</i>	<i>As specified in Table 2.1.4 for the applicable array/class</i>	<i>As specified in Table 2.1.4 for the applicable array/class</i>	<i>As specified in Table 2.1.4 for array/class 6x6B</i>	<i>As specified in Table 2.1.4 for array/class 6x6B</i>
<i>Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTU.</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTU.</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTIHM.</i>	<i>Cooling time <math>\geq</math> 18 years and average burnup <math>\leq</math> 30,000 MWD/MTIHM.</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i><math>\leq</math> 115 Watts</i>	<i><math>\leq</math> 115 Watts</i>	<i><math>\leq</math> 115 Watts</i>	<i><math>\leq</math> 115 Watts</i>
<i>Fuel Assembly Length</i>	<i><math>\leq</math> 135.0 in. (nominal design)</i>	<i><math>\leq</math> 135.0 in. (nominal design)</i>	<i><math>\leq</math> 135.0 in. (nominal design)</i>	<i><math>\leq</math> 135.0 in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq</math> 4.70 in. (nominal design)</i>	<i><math>\leq</math> 4.70 in. (nominal design)</i>	<i><math>\leq</math> 4.70 in. (nominal design)</i>	<i><math>\leq</math> 4.70 in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq</math> 400 lbs, (including channels)</i>	<i><math>\leq</math> 550 lbs, (including channels and DFC)</i>	<i><math>\leq</math> 400 lbs, (including channels)</i>	<i><math>\leq</math> 550 lbs, (including channels and DFC)</i>

Table 2.1.19 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

<b>PARAMETER</b>	<b>VALUE</b>
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity is limited to up to four (4) DFCs containing Dresden Unit 1 or Humboldt Bay uranium oxide or MOX fuel debris. The remaining fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:</i> <ul style="list-style-type: none"> <li>- <i>uranium oxide BWR intact fuel assemblies</i></li> <li>- <i>MOX BWR intact fuel assemblies</i></li> <li>- <i>uranium oxide BWR damaged fuel assemblies in DFCs</i></li> <li>- <i>MOX BWR damaged fuel assemblies in DFCs</i></li> <li>- <i>up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12.</i></li> </ul> </li> <li>▪ <i>Stainless steel channels are not permitted.</i></li> <li>▪ <i>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.</i></li> </ul>

Notes:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*
2. *Only fuel from the Dresden Unit 1 and Humboldt Bay plants are permitted for storage in the MPC-68F.*



Table 2.1.20

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>	
<i>Fuel Type</i>	<i>Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class</i>	<i>Uranium oxide PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>
<i>Maximum Initial Enrichment per Assembly</i>	<i>As specified in Table 2.1.3 for the applicable array/class</i>	<i>As specified in Table 2.1.3 for the applicable array/class</i>
<i>Post-irradiation Cooling Time, and Average Burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 8</math> yrs and <math>\leq 40,000</math> MWD/MTU</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 8</math> yrs and <math>\leq 40,000</math> MWD/MTU</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 710</math> Watts</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 710</math> Watts</i>
<i>Non-fuel hardware post-irradiation Cooling Time and Burnup</i>	<i>As specified in Table 2.1.25</i>	<i>As specified in Table 2.1.25</i>
<i>Fuel Assembly Length</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq 1680</math> lbs (including non-fuel hardware)</i>	<i><math>\leq 1680</math> lbs (including DFC and non-fuel hardware)</i>

Table 2.1.20 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E

<b>PARAMETER</b>	<b>VALUE</b>
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity is limited to up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies.</i></li> <li>▪ <i>Fuel debris and neutron sources are not authorized for storage in the MPC-24E.</i></li> <li>▪ <i>BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</i></li> <li>▪ <i>CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16.</i></li> <li>▪ <i>Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.</i></li> </ul>

Notes:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*

Table 2.1.21

LIMITS FOR MATERIAL TO BE STORED IN MPC-32

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>	
<i>Fuel Type</i>	<i>Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.</i>	<i>Uranium oxide, PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>
<i>Maximum Initial Enrichment per Assembly</i>	<i>As specified in Table 2.1.3 for the applicable fuel assembly array/class</i>	<i>As specified in Table 2.1.3 for the applicable fuel assembly array/class</i>
<i>Post-irradiation Cooling Time and Average Burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 9</math> years and <math>\leq 30,000</math> MWD/MTU or <math>\geq 20</math> years and <math>\leq 40,000</math> MWD/MTU</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 9</math> years and <math>\leq 30,000</math> MWD/MTU or <math>\geq 20</math> years and <math>\leq 40,000</math> MWD/MTU</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR-clad: As specified in Section 2.1.9.1</i>  <i>SS-clad: <math>\leq 500</math> Watts</i>	<i>ZR-clad: As specified in Section 2.1.9.1</i>  <i>SS-clad: <math>\leq 500</math> Watts</i>
<i>Non-fuel hardware post-irradiation cooling time and burnup</i>	<i>As specified in Table 2.1.25</i>	<i>As specified in Table 2.1.25</i>
<i>Fuel Assembly Length</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq 1,680</math> lbs (including non-fuel hardware)</i>	<i><math>\leq 1,680</math> lbs (including DFC and non-fuel hardware)</i>

Table 2.1.21 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32

<b>PARAMETER</b>	<b>VALUE</b>
<i>Other Limits</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32.</i></li> <li>▪ <i>Fuel debris and neutron sources are not permitted for storage in MPC-32.</i></li> <li>▪ <i>BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</i></li> <li>▪ <i>CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20.</i></li> <li>▪ <i>Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.</i></li> </ul>

NOTES:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*

Table 2.1.22

LIMITS FOR MATERIAL TO BE STORED IN MPC-68FF

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>	
<i>Fuel Type</i>	<i>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.</i>	<i>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.</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class</i>
<i>Maximum Initial Planar Average Enrichment per Assembly and Rod Enrichment</i>	<i>As specified in Table 2.1.4 for the applicable fuel assembly array/class</i>	<i>Planar Average:</i>  <i>≤ 2.7 wt% <sup>235</sup>U for array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A;</i>  <i>≤ 4.0 wt% <sup>235</sup>U for all other array/classes</i>  <i>Rod:</i>  <i>As specified in Table 2.1.4</i>
<i>Post-irradiation cooling time and average burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: Note 4</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: Note 4.</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: ≤ 95 Watts</i>	<i>ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.</i>  <i>SS clad: ≤ 95 Watts</i>
<i>Fuel Assembly Length</i>	<i>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design)</i>  <i>All Other array/classes: ≤ 176.5 in. (nominal design)</i>	<i>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design)</i>  <i>All Other array/classes: ≤ 176.5 in. (nominal design)</i>

Table 2.1.22 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Assembly Width	<p>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: <math>\leq 4.7</math> in. (nominal design)</p> <p>All Other array/classes: <math>\leq 5.85</math> in. (nominal design)</p>	<p>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: <math>\leq 4.7</math> in. (nominal design)</p> <p>All Other array/classes: <math>\leq 5.85</math> in. (nominal design)</p>
Fuel Assembly Weight	<p>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: <math>\leq 550</math> lbs. (including channels)</p> <p>All Other array/classes: <math>\leq 700</math> lbs. (including channels)</p>	<p>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: <math>\leq 550</math> lbs. (including channels and DFC)</p> <p>All Other array/classes: <math>\leq 700</math> lbs. (including channels and DFC)</p>
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to up to one (1) Up to eight (8) Dresden Unit 1 or Humboldt Bay fuel assemblies classified as fuel debris in DFCs, and any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in DFCs and intact fuel assemblies up to a total of 68.</li> <li>▪ Up to 16 damaged fuel assemblies and/or up to eight (8) fuel assemblies classified as fuel debris from plants other than Dresden Unit 1 or Humboldt Bay may be stored in DFCs in MPC-68FF. 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.</li> <li>▪ 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.</li> <li>▪ 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.</li> </ul>	

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  $\geq 18$  years, an average burnup  $\leq 30,000$  MWD/MTU, and a maximum decay heat  $\leq 115$  Watts.
3. Array/class 8x8F fuel assemblies shall have a cooling time  $\geq 10$  years, an average burnup  $\leq 27,500$  MWD/MTU, and a maximum decay  $\leq 183.5$  Watts.
4. SS-clad fuel assemblies shall have a cooling time  $\geq 10$  years, and an average burnup  $\leq 22,500$  MWD/MTU.

Table 2.1.23

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>	
<i>Fuel Type</i>	<i>Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class</i>	<i>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)</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>	<i>ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class</i>
<i>Maximum Initial Enrichment per Assembly</i>	<i>As specified in Table 2.1.3 for the applicable array/class</i>	<i>As specified in Table 2.1.3 for the applicable array/class</i>
<i>Post-irradiation Cooling Time, and Average Burnup per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 8</math> yrs and <math>\leq 40,000</math> MWD/MTU</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 8</math> yrs and <math>\leq 40,000</math> MWD/MTU</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 710</math> Watts</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 710</math> Watts</i>
<i>Non-fuel hardware post-irradiation Cooling Time and Burnup</i>	<i>As specified in Table 2.1.25</i>	<i>As specified in Table 2.1.25</i>
<i>Fuel Assembly Length</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq 1680</math> lbs (including non-fuel hardware)</i>	<i><math>\leq 1680</math> lbs (including DFC and non-fuel hardware)</i>

Table 2.1.23 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

<b>PARAMETER</b>	<b>VALUE</b>
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity per MPC: up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies.</i></li> <li>▪ <i>Neutron sources are not authorized for storage in the MPC-24EF.</i></li> <li>▪ <i>BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</i></li> <li>▪ <i>CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16.</i></li> <li>▪ <i>Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.</i></li> </ul>

Notes:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*



Table 2.1.24

LIMITS FOR MATERIAL TO BE STORED IN MPC-32F

<b>PARAMETER</b>	<b>VALUE (Note 1)</b>	
<i>Fuel Type</i>	<i>Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class</i>	<i>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</i>
<i>Cladding Type</i>	<i>ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class</i>	<i>ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class</i>
<i>Maximum Initial Enrichment per Assembly</i>	<i>As specified in Table 2.1.3</i>	<i>As specified in Table 2.1.3</i>
<i>Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 9</math> years and <math>\leq 30,000</math> MWD/MTU or <math>\geq 20</math> years and <math>\leq 40,000</math> MWD/MTU</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\geq 9</math> years and <math>\leq 30,000</math> MWD/MTU or <math>\geq 20</math> years and <math>\leq 40,000</math> MWD/MTU</i>
<i>Decay Heat Per Fuel Storage Location</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 500</math> Watts</i>	<i>ZR clad: As specified in Section 2.1.9.1</i>  <i>SS clad: <math>\leq 500</math> Watts</i>
<i>Non-fuel hardware post-irradiation Cooling Time and Burnup</i>	<i>As specified in Table 2.1.25</i>	<i>As specified in Table 2.1.25</i>
<i>Fuel Assembly Length</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>	<i><math>\leq 176.8</math> in. (nominal design)</i>
<i>Fuel Assembly Width</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>	<i><math>\leq 8.54</math> in. (nominal design)</i>
<i>Fuel Assembly Weight</i>	<i><math>\leq 1,680</math> lbs (including non-fuel hardware)</i>	<i><math>\leq 1,680</math> lbs (including DFC and non-fuel hardware)</i>

Table 2.1.24 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32F

<b>PARAMETER</b>	<b>VALUE</b>
<i>Other Limitations</i>	<ul style="list-style-type: none"> <li>▪ <i>Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32.</i></li> <li>▪ <i>Neutron sources are not permitted for storage in MPC-32.</i></li> <li>▪ <i>BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</i></li> <li>▪ <i>CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20.</i></li> <li>▪ <i>Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.</i></li> </ul>

NOTES:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*

Table 2.1.25

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

<b>Post-irradiation Cooling Time (yrs)</b>	<b>Inserts (Note 4) Maximum Burnup (MWD/MTU)</b>	<b>Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)</b>	<b>Control Component (Note 6) Maximum Burnup (MWD/MTU)</b>	<b>APSR Maximum Burnup (MWD/MTU)</b>
≥ 3	≤ 24,635	N/A (Note 7)	N/A	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	-	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 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 TPD 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.
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
7. N/A means not authorized for loading at this cooling time.

Table 2.1.26

MAXIMUM ALLOWABLE DECAY HEAT PER FUEL STORAGE LOCATION  
(UNIFORM LOADING, ZR-CLAD)

<i>MPC Model</i>	<i>Decay Heat per Fuel Assembly (kW)</i>
<i>Intact Fuel Assemblies</i>	
<i>MPC-24</i>	$\leq 1.157$
<i>MPC-24E/24EF</i>	$\leq 1.173$
<i>MPC-32/32F</i>	$\leq 0.898$
<i>MPC-68/68FF</i>	$\leq 0.414$
<i>Damaged Fuel Assemblies and Fuel Debris</i>	
<i>MPC-24</i>	$\leq 1.099$
<i>MPC-24E/24EF</i>	$\leq 1.114$
<i>MPC-32/32F</i>	$\leq 0.718$
<i>MPC-68/68FF</i>	$\leq 0.393$

Table 2.1.27

*MPC FUEL STORAGE REGIONS AND MAXIMUM DECAY HEAT*

<i>MPC Model</i>	<i>Number of Fuel Storage Locations in Inner and Outer Regions</i>	<i>Inner Region Maximum Decay Heat per Assembly (kW)</i>	<i>Outer Region Maximum Decay Heat per Assembly (kW)</i>
<i>MPC-24</i>	<i>4 and 20</i>	<i>1.470</i>	<i>0.900</i>
<i>MPC-24E/24EF</i>	<i>4 and 20</i>	<i>1.540</i>	<i>0.900</i>
<i>MPC-32/32F</i>	<i>12 and 20</i>	<i>1.131</i>	<i>0.600</i>
<i>MPC-68/68FF</i>	<i>32 and 36</i>	<i>0.500</i>	<i>0.275</i>

*Note: These limits apply to intact fuel assemblies, damaged fuel assemblies and fuel debris.*

Table 2.1.28

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

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
≥ 3	20277.1	303.592	-68.329	-139.41	2993.67	-498.159	-615.411
≥ 4	35560.1	-6034.67	985.415	-132.734	3578.92	-723.721	-609.84
≥ 5	48917.9	-14499.5	2976.09	-150.707	4072.55	-892.691	-54.8362
≥ 6	59110.3	-22507	5255.61	-177.017	4517.03	-1024.01	613.36
≥ 7	67595.6	-30158.1	7746.6	-200.128	4898.71	-1123.21	716.004
≥ 8	74424.9	-36871.1	10169.4	-218.676	5203.64	-1190.24	741.163
≥ 9	81405.8	-44093.1	12910.8	-227.916	5405.34	-1223.27	250.224
≥ 10	86184.3	-49211.7	15063.4	-237.641	5607.96	-1266.21	134.435
≥ 11	92024.9	-55666.8	17779.6	-240.973	5732.25	-1282.12	-401.456
≥ 12	94775.8	-58559.7	19249.9	-246.369	5896.27	-1345.42	-295.435
≥ 13	100163	-64813.8	22045.1	-242.572	5861.86	-1261.66	-842.159
≥ 14	103971	-69171	24207	-242.651	5933.96	-1277.48	-1108.99
≥ 15	108919	-75171.1	27152.4	-243.154	6000.2	-1301.19	-1620.63
≥ 16	110622	-76715.2	28210.2	-240.235	6028.33	-1307.74	-1425.5
≥ 17	115582	-82929.7	31411.9	-235.234	5982.3	-1244.11	-1948.05
≥ 18	119195	-87323.5	33881.4	-233.28	6002.43	-1245.95	-2199.41
≥ 19	121882	-90270.6	35713.7	-231.873	6044.42	-1284.55	-2264.05
≥ 20	124649	-93573.5	37853.1	-230.22	6075.82	-1306.57	-2319.63

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	18937.9	70.2997	-28.6224	-130.732	2572.36	-383.393	-858.17
≥ 4	32058.7	-4960.63	745.224	-125.978	3048.98	-551.656	-549.108
≥ 5	42626.3	-10804.1	1965.09	-139.722	3433.49	-676.643	321.88
≥ 6	51209.6	-16782.3	3490.45	-158.929	3751.01	-761.524	847.282
≥ 7	57829.9	-21982	5009.12	-180.026	4066.65	-846.272	1200.45
≥ 8	62758	-26055.3	6330.88	-196.804	4340.18	-928.336	1413.17
≥ 9	68161.4	-30827.6	7943.87	-204.454	4500.52	-966.347	1084.69
≥ 10	71996.8	-34224.3	9197.25	-210.433	4638.94	-1001.83	1016.38
≥ 11	75567.3	-37486.1	10466.9	-214.95	4759.55	-1040.85	848.169
≥ 12	79296.7	-40900.3	11799.6	-212.898	4794.13	-1040.51	576.242
≥ 13	82257.3	-43594	12935	-212.8	4845.81	-1056.01	410.807
≥ 14	83941.2	-44915.2	13641	-215.389	4953.19	-1121.71	552.724
≥ 15	87228.5	-48130	15056.9	-212.545	4951.12	-1112.5	260.194
≥ 16	90321.7	-50918.3	16285.5	-206.094	4923.36	-1106.35	-38.7487
≥ 17	92836.2	-53314.5	17481.7	-203.139	4924.61	-1109.32	-159.673
≥ 18	93872.8	-53721.4	17865.1	-202.573	4956.21	-1136.9	30.0594
≥ 19	96361.6	-56019.1	19075.9	-199.068	4954.59	-1156.07	-125.917
≥ 20	98647.5	-57795.1	19961.8	-191.502	4869.59	-1108.74	-217.603

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	19176.9	192.012	-66.7595	-138.112	2666.73	-407.664	-1372.41
≥ 4	32040.3	-4731.4	651.014	-124.944	3012.63	-530.456	-890.059
≥ 5	43276.7	-11292.8	2009.76	-142.172	3313.91	-594.917	-200.195
≥ 6	51315.5	-16920.5	3414.76	-164.287	3610.77	-652.118	463.041
≥ 7	57594.7	-21897.6	4848.49	-189.606	3940.67	-729.367	781.46
≥ 8	63252.3	-26562.8	6273.01	-199.974	4088.41	-732.054	693.879
≥ 9	67657.5	-30350.9	7533.4	-211.77	4283.39	-772.916	588.456
≥ 10	71834.4	-34113.7	8857.32	-216.408	4383.45	-774.982	380.243
≥ 11	75464.1	-37382.1	10063	-218.813	4460.69	-776.665	160.668
≥ 12	77811.1	-39425.1	10934.3	-225.193	4604.68	-833.459	182.463
≥ 13	81438.3	-42785.4	12239.9	-220.943	4597.28	-803.32	-191.636
≥ 14	84222.1	-45291.6	13287.9	-218.366	4608.13	-791.655	-354.59
≥ 15	86700.1	-47582.6	14331.2	-218.206	4655.34	-807.366	-487.316
≥ 16	88104.7	-48601.1	14927.9	-219.498	4729.97	-849.446	-373.196
≥ 17	91103.3	-51332.5	16129	-212.138	4679.91	-822.896	-654.296
≥ 18	93850.4	-53915.8	17336.9	-207.666	4652.65	-799.697	-866.307
≥ 19	96192.9	-55955.8	18359.3	-203.462	4642.65	-800.315	-1007.75
≥ 20	97790.4	-57058.1	19027.7	-200.963	4635.88	-799.721	-951.122



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	15789.2	119.829	-21.8071	-127.422	2152.53	-267.717	-580.768
≥ 4	26803.8	-3312.93	415.027	-116.279	2550.15	-386.33	-367.168
≥ 5	36403.6	-7831.93	1219.66	-126.065	2858.32	-471.785	326.863
≥ 6	44046.1	-12375.9	2213.52	-145.727	3153.45	-539.715	851.971
≥ 7	49753.5	-16172.6	3163.61	-166.946	3428.38	-603.598	1186.31
≥ 8	55095.4	-20182.5	4287.03	-183.047	3650.42	-652.92	1052.4
≥ 9	58974.4	-23071.6	5156.53	-191.718	3805.41	-687.18	1025
≥ 10	62591.8	-25800.8	5995.95	-195.105	3884.14	-690.659	868.556
≥ 11	65133.1	-27747.4	6689	-203.095	4036.91	-744.034	894.607
≥ 12	68448.4	-30456	7624.9	-202.201	4083.52	-753.391	577.914
≥ 13	71084.4	-32536.4	8381.78	-201.624	4117.93	-757.16	379.105
≥ 14	73459.5	-34352.3	9068.86	-197.988	4113.16	-747.015	266.536
≥ 15	75950.7	-36469.4	9920.52	-199.791	4184.91	-779.222	57.9429
≥ 16	76929.1	-36845.6	10171.3	-197.88	4206.24	-794.541	256.099
≥ 17	79730	-39134.8	11069.4	-190.865	4160.42	-773.448	-42.6853
≥ 18	81649.2	-40583	11736.1	-187.604	4163.36	-785.838	-113.614
≥ 19	83459	-41771.8	12265.9	-181.461	4107.51	-758.496	-193.442
≥ 20	86165.4	-44208.8	13361.2	-178.89	4107.62	-768.671	-479.778

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						
	A	B	C	D	E	F	G
≥ 3	15192.5	50.5722	-12.3042	-126.906	2009.71	-235.879	-561.574
≥ 4	25782.5	-3096.5	369.096	-113.289	2357.75	-334.695	-254.964
≥ 5	35026.5	-7299.87	1091.93	-124.619	2664	-414.527	470.916
≥ 6	42234.9	-11438.4	1967.63	-145.948	2945.81	-474.981	1016.84
≥ 7	47818.4	-15047	2839.22	-167.273	3208.95	-531.296	1321.12
≥ 8	52730.7	-18387.2	3702.43	-175.057	3335.58	-543.232	1223.61
≥ 9	56254.6	-20999.9	4485.93	-190.489	3547.98	-600.64	1261.55
≥ 10	59874.6	-23706.5	5303.88	-193.807	3633.01	-611.892	1028.63
≥ 11	62811	-25848.4	5979.64	-194.997	3694.14	-618.968	862.738
≥ 12	65557.6	-27952.4	6686.74	-198.224	3767.28	-635.126	645.139
≥ 13	67379.4	-29239.2	7197.49	-200.164	3858.53	-677.958	652.601
≥ 14	69599.2	-30823.8	7768.51	-196.788	3868.2	-679.88	504.443
≥ 15	71806.7	-32425	8360.38	-191.935	3851.65	-669.917	321.146
≥ 16	73662.6	-33703.5	8870.78	-187.366	3831.59	-658.419	232.335
≥ 17	76219.8	-35898.1	9754.72	-189.111	3892.07	-694.244	-46.924
≥ 18	76594.4	-35518.2	9719.78	-185.11	3897.04	-712.82	236.047
≥ 19	78592.7	-36920.8	10316.5	-179.54	3865.84	-709.551	82.478
≥ 20	80770.5	-38599.9	11051.3	-175.106	3858.67	-723.211	-116.014

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	17038.2	158.445	-37.6008	-136.707	2368.1	-321.58	-700.033
≥ 4	29166.3	-3919.95	508.439	-125.131	2782.53	-455.722	-344.199
≥ 5	40285	-9762.36	1629.72	-139.652	3111.83	-539.804	139.67
≥ 6	48335.7	-15002.6	2864.09	-164.702	3444.97	-614.756	851.706
≥ 7	55274.9	-20190	4258.03	-185.909	3728.11	-670.841	920.035
≥ 8	60646.6	-24402.4	5483.54	-199.014	3903.29	-682.26	944.913
≥ 9	64663.2	-27753.1	6588.21	-215.318	4145.34	-746.822	967.914
≥ 10	69306.9	-31739.1	7892.13	-218.898	4237.04	-746.815	589.277
≥ 11	72725.8	-34676.6	8942.26	-220.836	4312.93	-750.85	407.133
≥ 12	76573.8	-38238.7	10248.1	-224.934	4395.85	-757.914	23.7549
≥ 13	78569	-39794.3	10914.9	-224.584	4457	-776.876	69.428
≥ 14	81559.4	-42453.6	11969.6	-222.704	4485.28	-778.427	-203.031
≥ 15	84108.6	-44680.4	12897.8	-218.387	4460	-746.756	-329.078
≥ 16	86512.2	-46766.8	13822.8	-216.278	4487.79	-759.882	-479.729
≥ 17	87526.7	-47326.2	14221	-218.894	4567.68	-805.659	-273.692
≥ 18	90340.3	-49888.6	15349.8	-212.139	4506.29	-762.236	-513.316
≥ 19	93218.2	-52436.7	16482.4	-207.653	4504.12	-776.489	-837.1
≥ 20	95533.9	-54474.1	17484.2	-203.094	4476.21	-760.482	-955.662

Table 2.1.28 (cont'd)

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

Cooling Time (years)	Array/Class 17x17A						
	A	B	C	D	E	F	G
≥ 3	16784.4	3.90244	-10.476	-128.835	2256.98	-287.108	-263.081
≥ 4	28859	-3824.72	491.016	-120.108	2737.65	-432.361	-113.457
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

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	15526.8	18.0364	-9.36581	-128.415	2050.81	-243.915	-426.07
≥ 4	26595.4	-3345.47	409.264	-115.394	2429.48	-350.883	-243.477
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953

Table 2.1.29

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

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

Table 2.1.29 (cont'd)

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

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

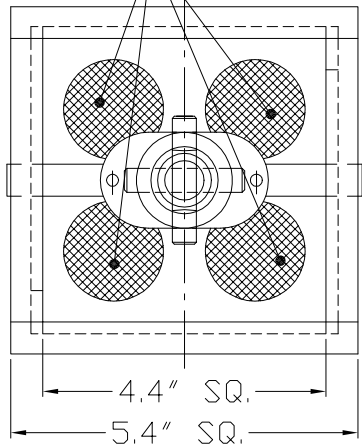
Cooling Time (years)	Array/Class 10x10A/B						
	A	B	C	D	E	F	G
≥ 3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥ 4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥ 5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥ 6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥ 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 2.1.29 (cont'd)

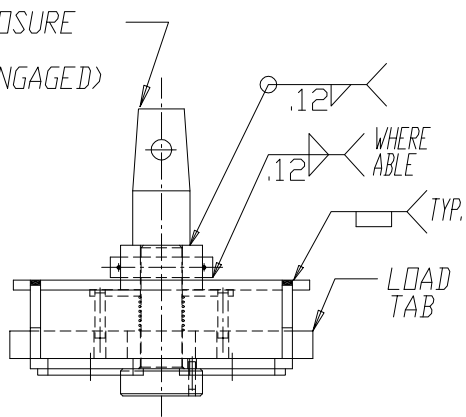
BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS  
(ZR-CLAD FUEL)

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

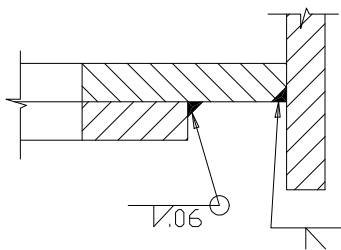
DEBRIS SCREEN



UPPER CLOSURE ASSEMBLY (SHOWN ENGAGED)



UPPER CLOSURE ASSEMBLY

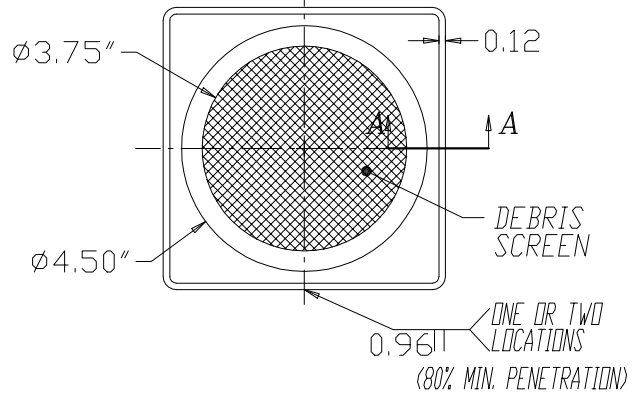
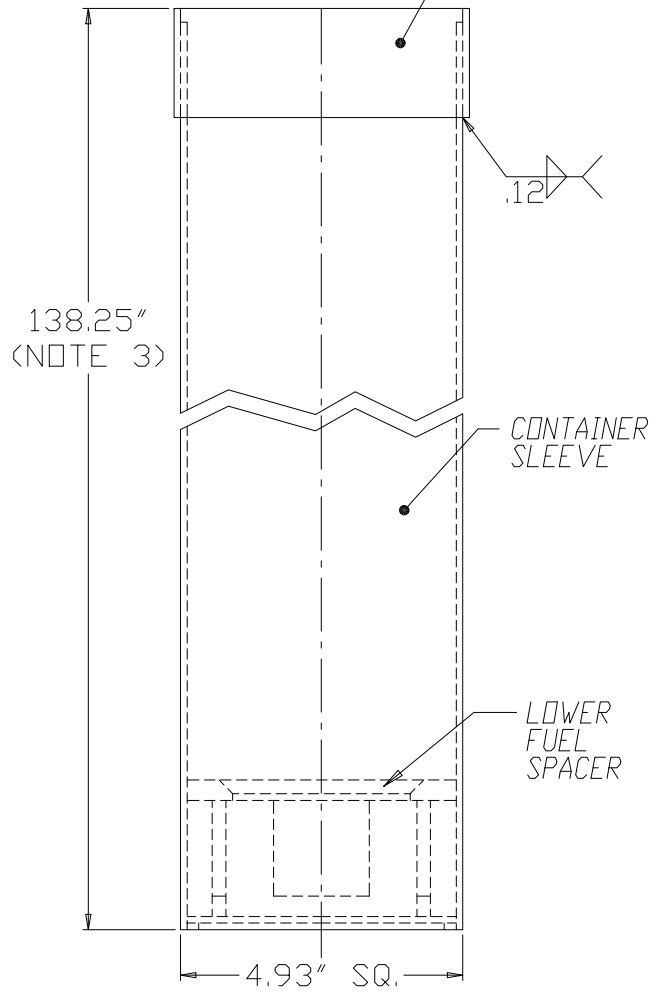


SECTION A-A

NOTE:

1. ALL DIMENSIONS ARE NOMINAL
2. ALL MATERIAL IS STAINLESS STEEL
3. LENGTH OF DFC VARIES WITH FUEL LENGTH.

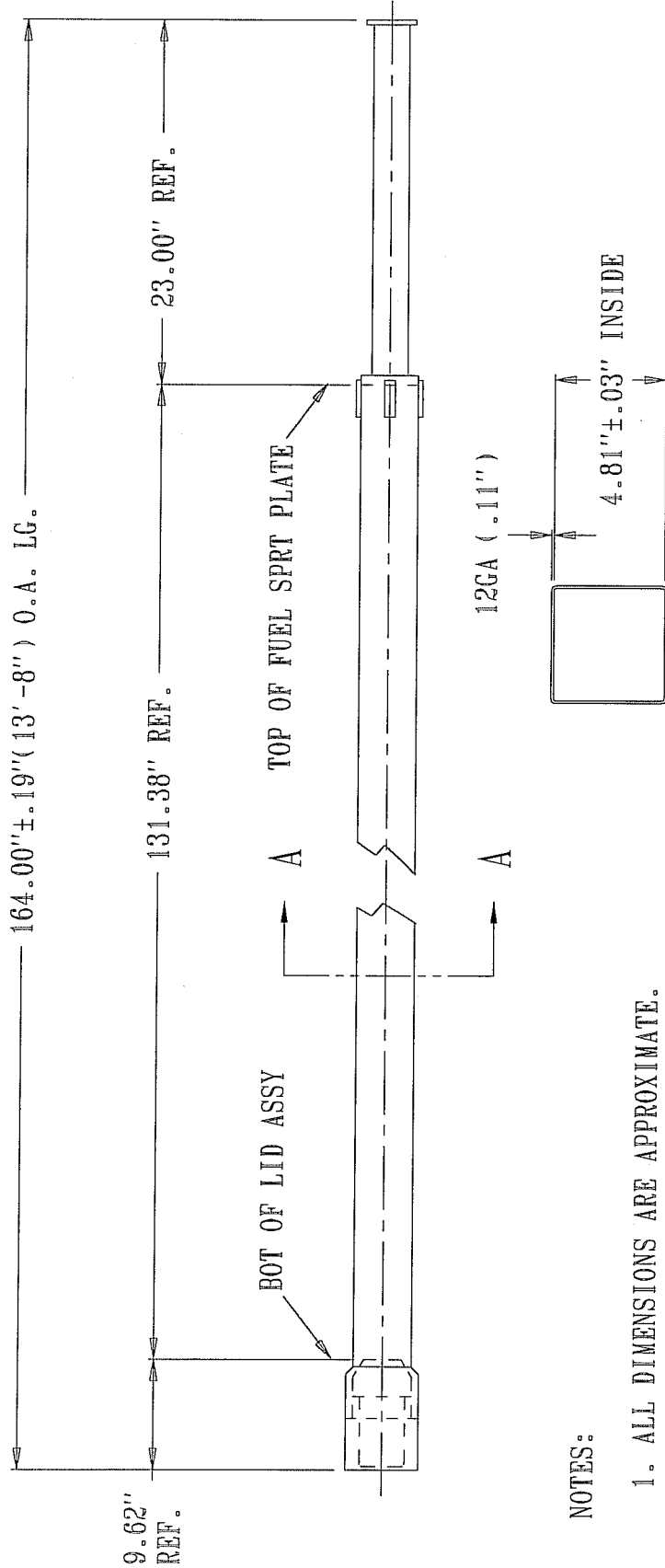
CONTAINER COLLAR



BOTTOM VIEW

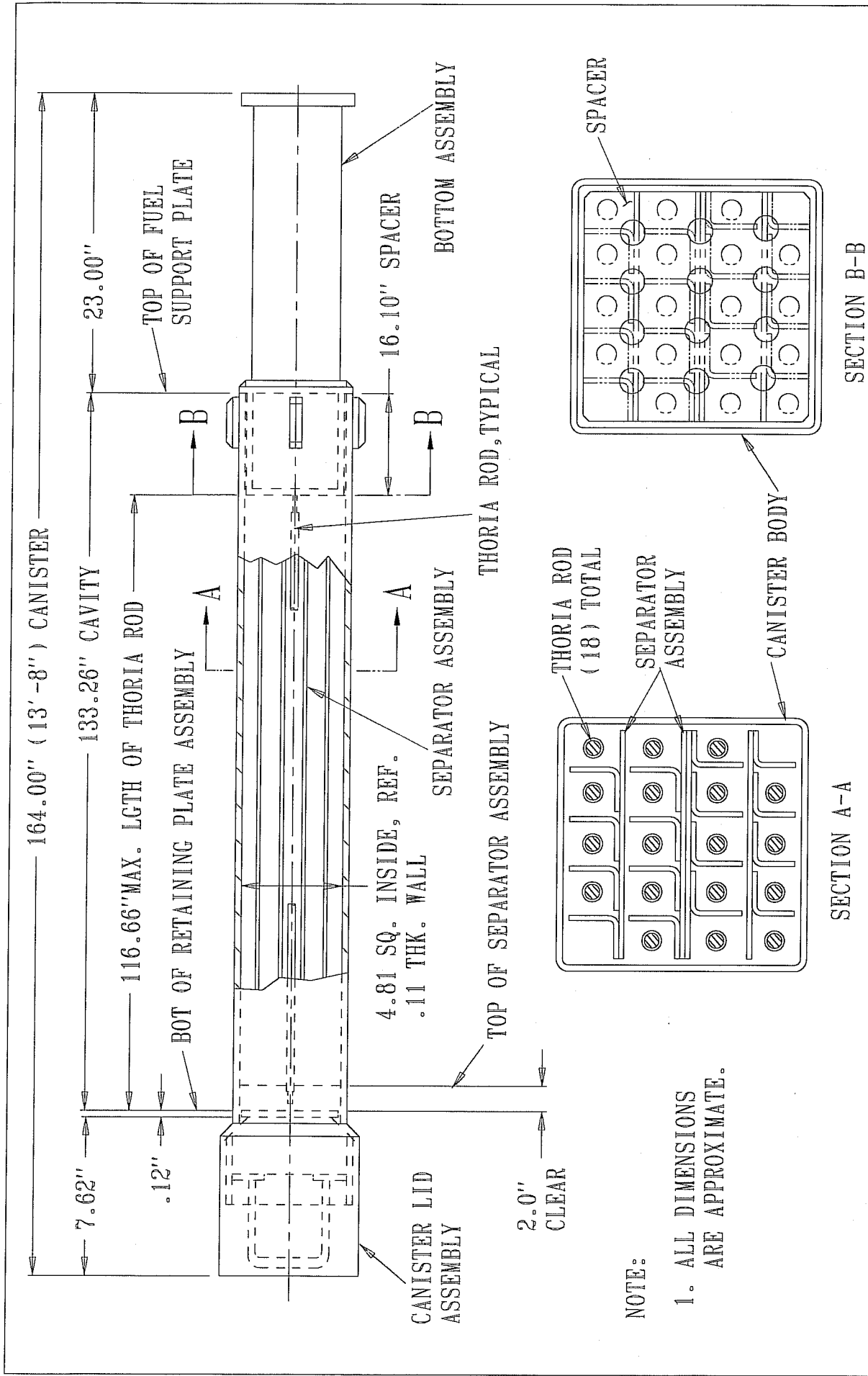
FIGURE 2.1.1; DAMAGED FUEL CONTAINER FOR DRESDEN UNIT-1/ HUMBOLDT BAY SNF





NOTES:  
 1. ALL DIMENSIONS ARE APPROXIMATE.

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



NOTE:  
1. ALL DIMENSIONS ARE APPROXIMATE.

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

- NOTES:  
 1. ALL DIMENSIONS ARE APPROXIMATE.  
 2. ALL MATERIAL IS STAINLESS STEEL.

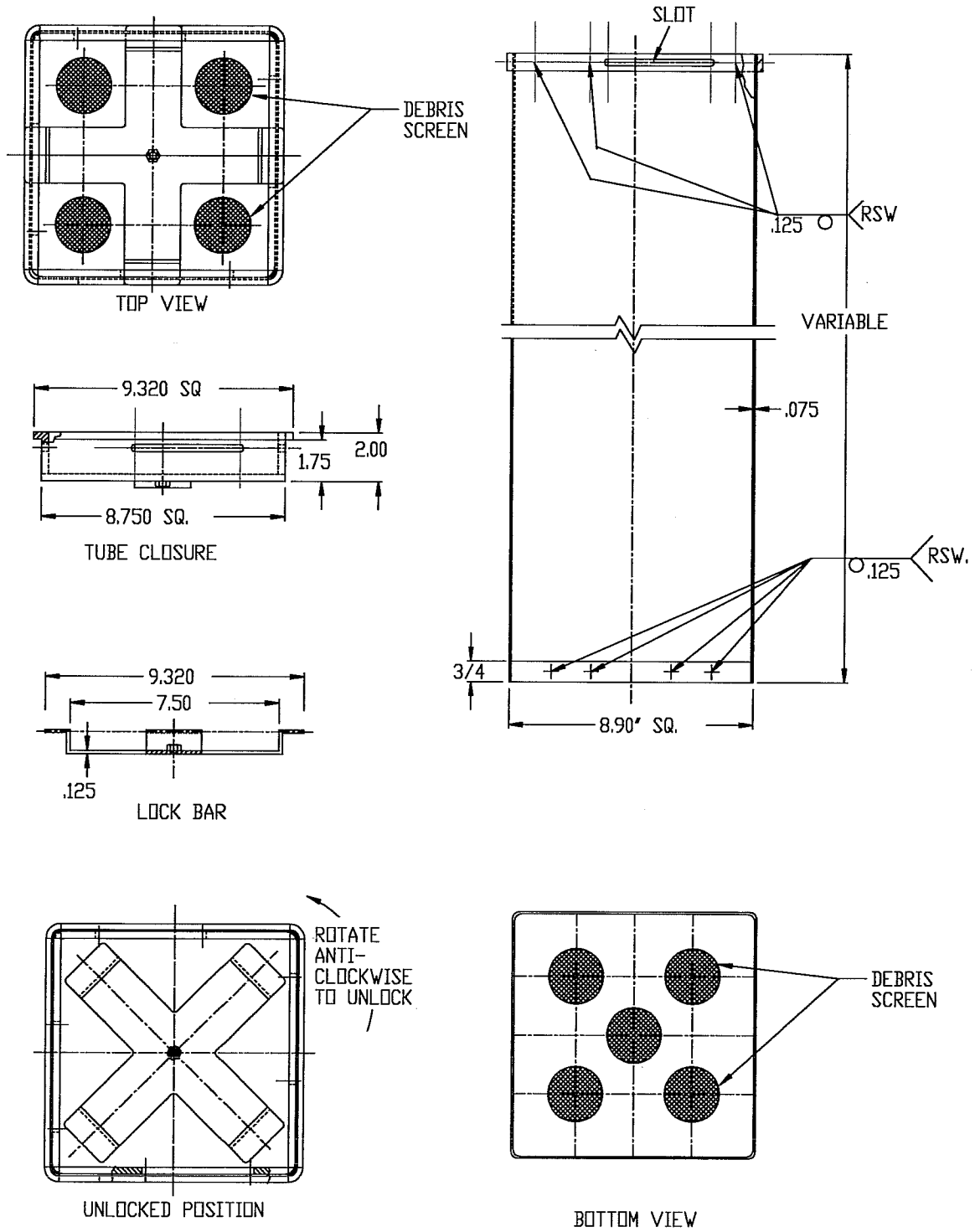
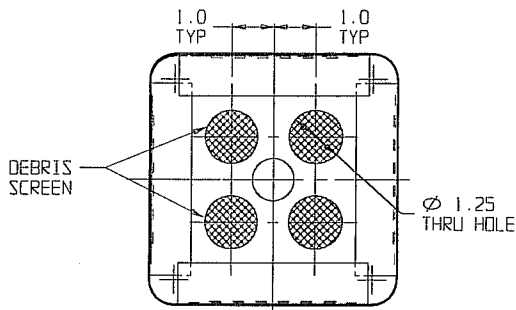
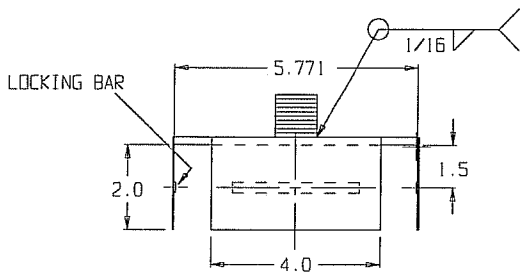


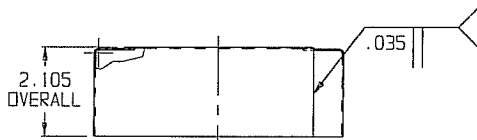
FIGURE 2.1.2B; HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-24E/24EF



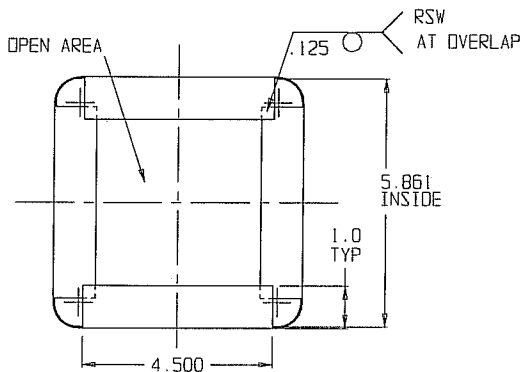
TUBE CAP AND WRAPPER PLAN



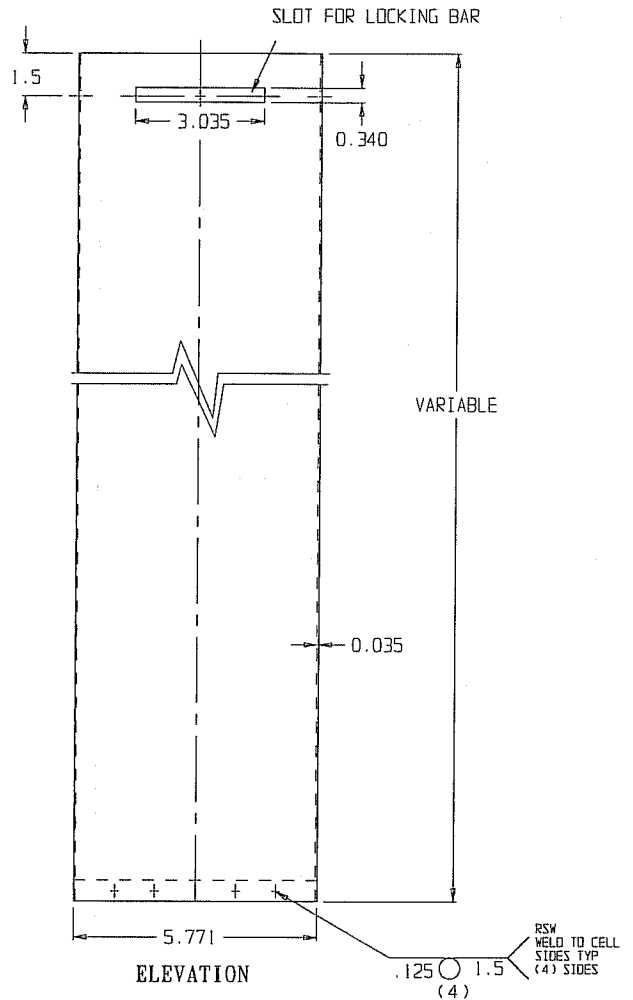
TUBE CAP ELEVATION



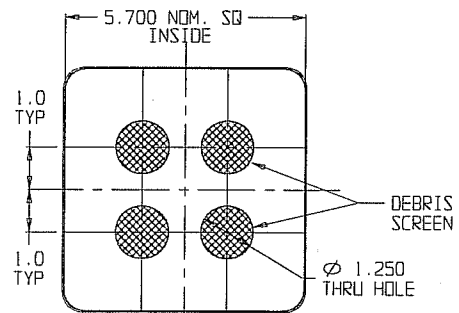
CAP WRAPPER ELEVATION



CAP WRAPPER PLAN



ELEVATION



BOTTOM VIEW

FIGURE 2.1.2C; HOLTEC DAMAGED FUEL CONTAINER FOR BWR SNF IN MPC-68/68FF

NOTES:  
 1. ALL DIMENSIONS ARE IN INCHES AND ARE APPROXIMATE.  
 2. ALL MATERIAL IS STAINLESS STEEL.

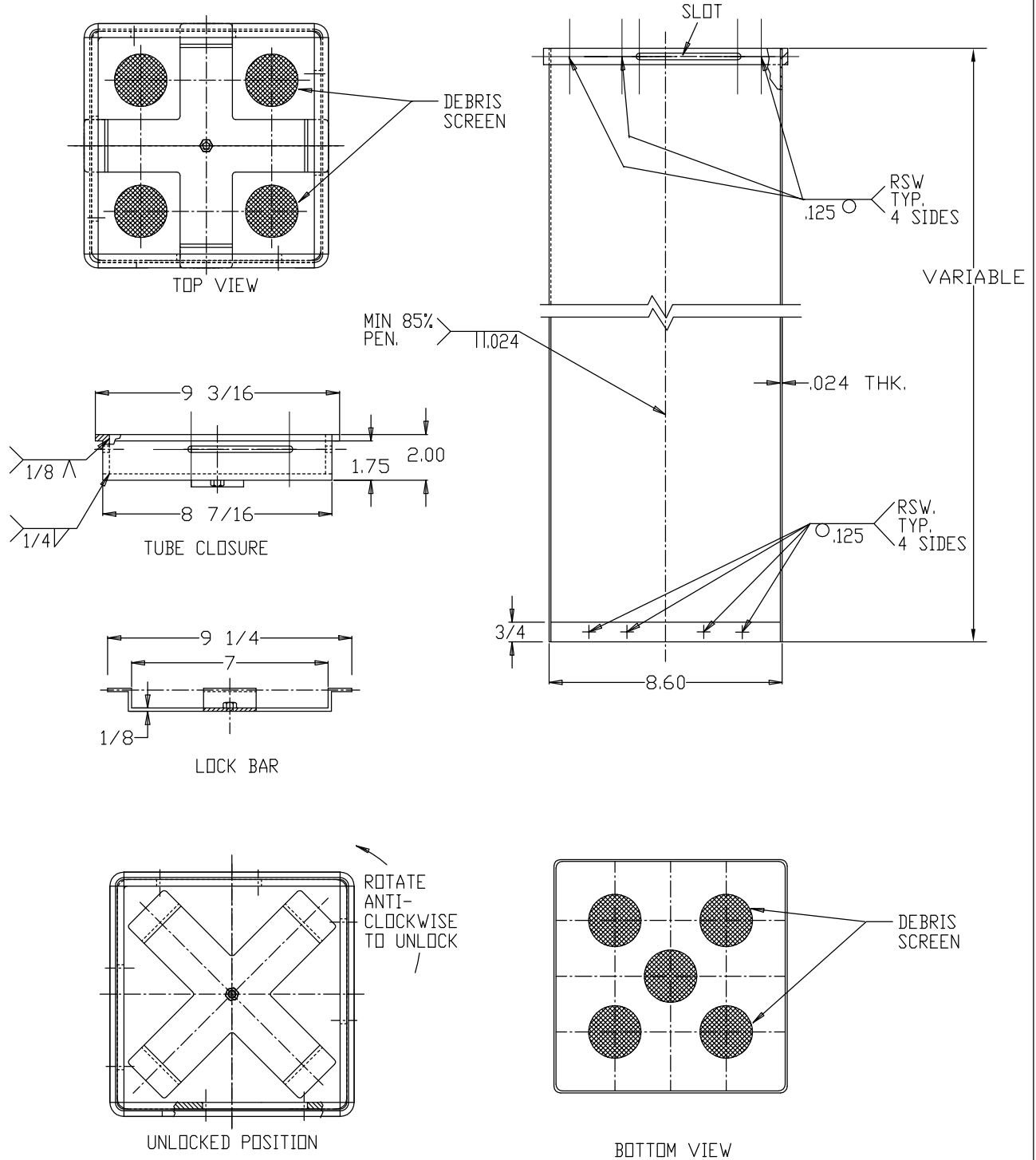


FIGURE 2.1.2D; HOLTEC DAMAGED FUEL CONTAINER FOR PWR SNF IN MPC-32/32F

### PWR Axial Burnup Distribution

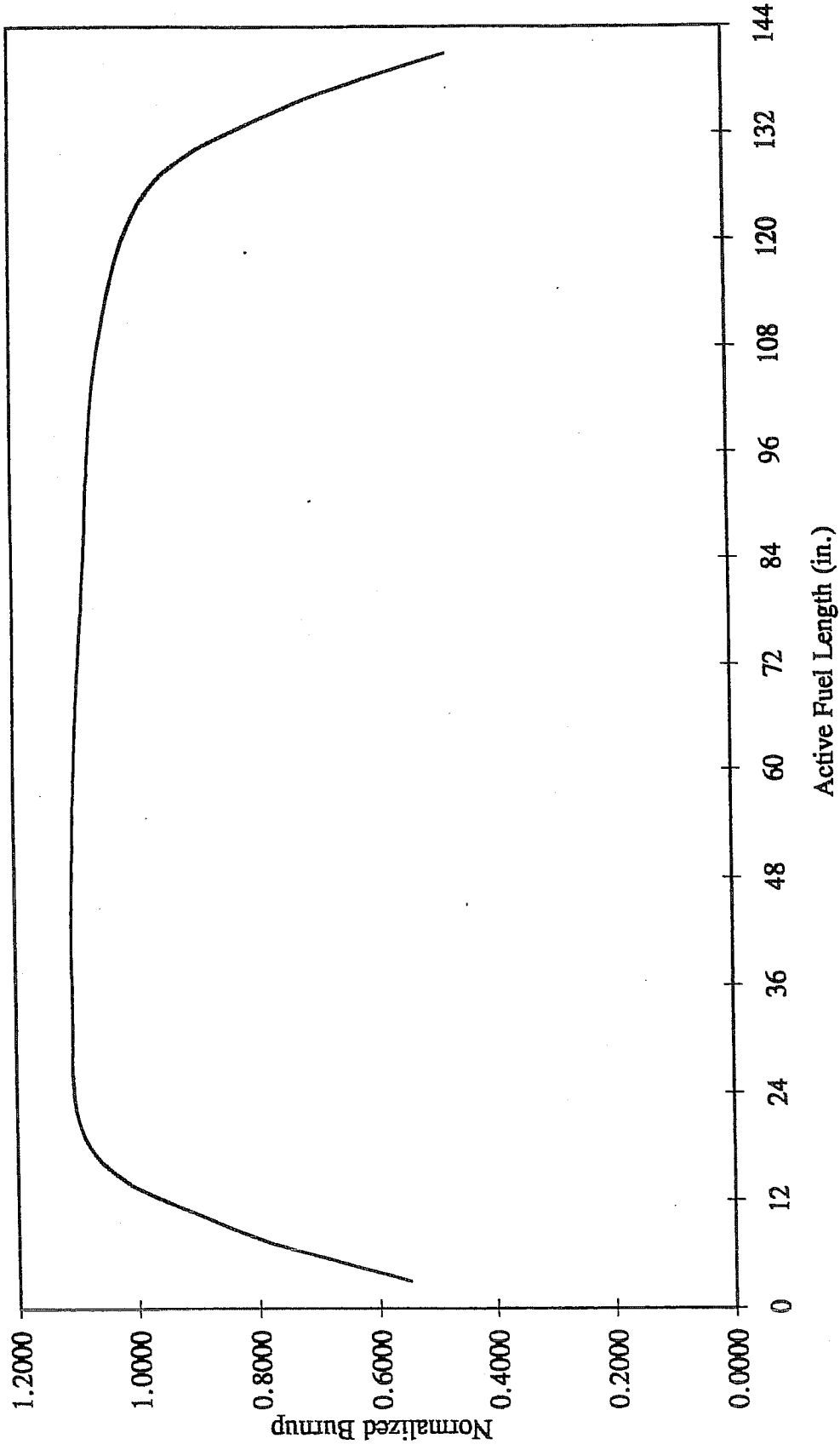


Figure 2.1.3; PWR Axial Burnup Profile with Normalized Distribution

### BWR Axial Burnup Distribution

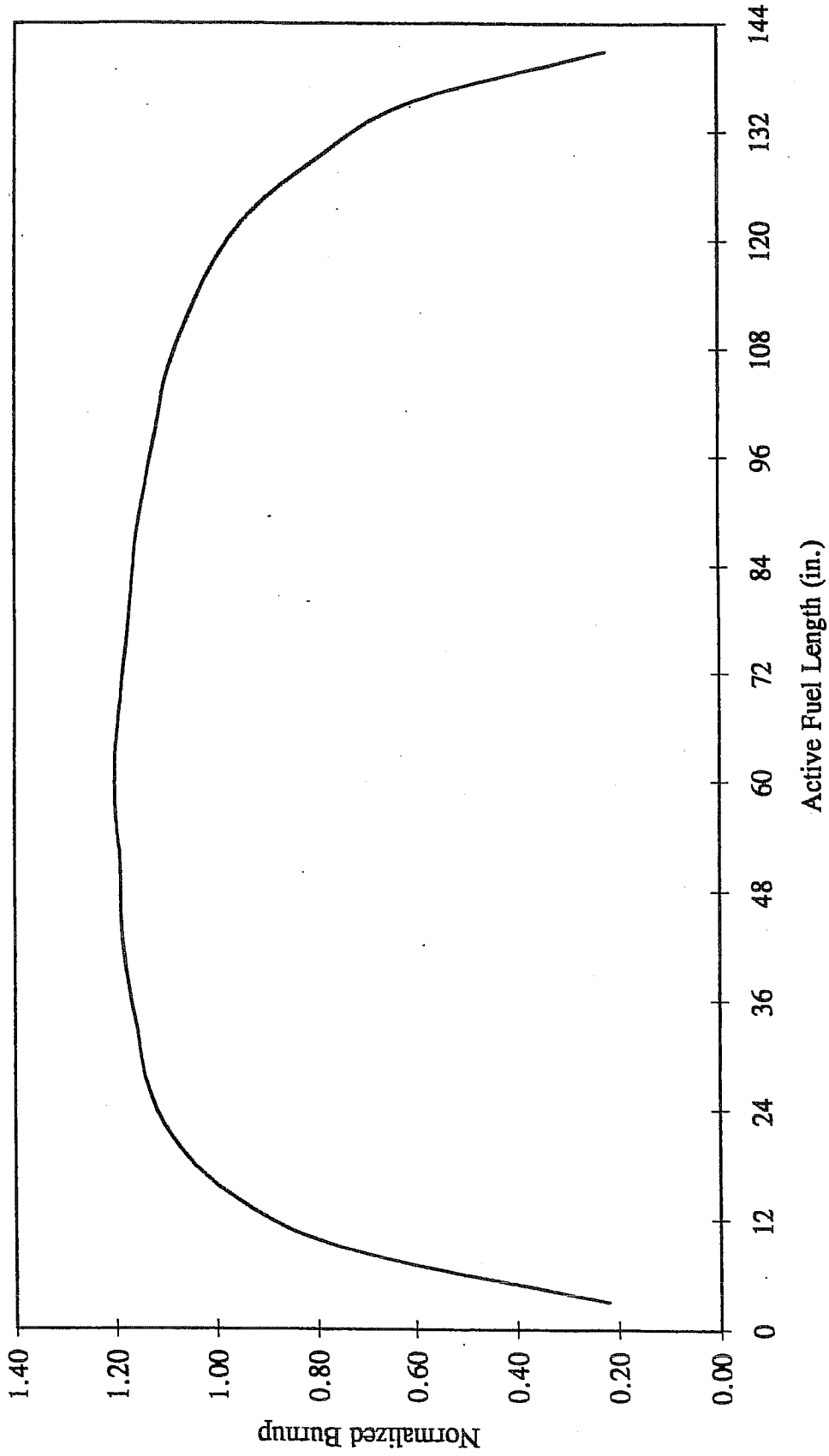


Figure 2.1.4; BWR Axial Burnup Profile with Normalized Distribution

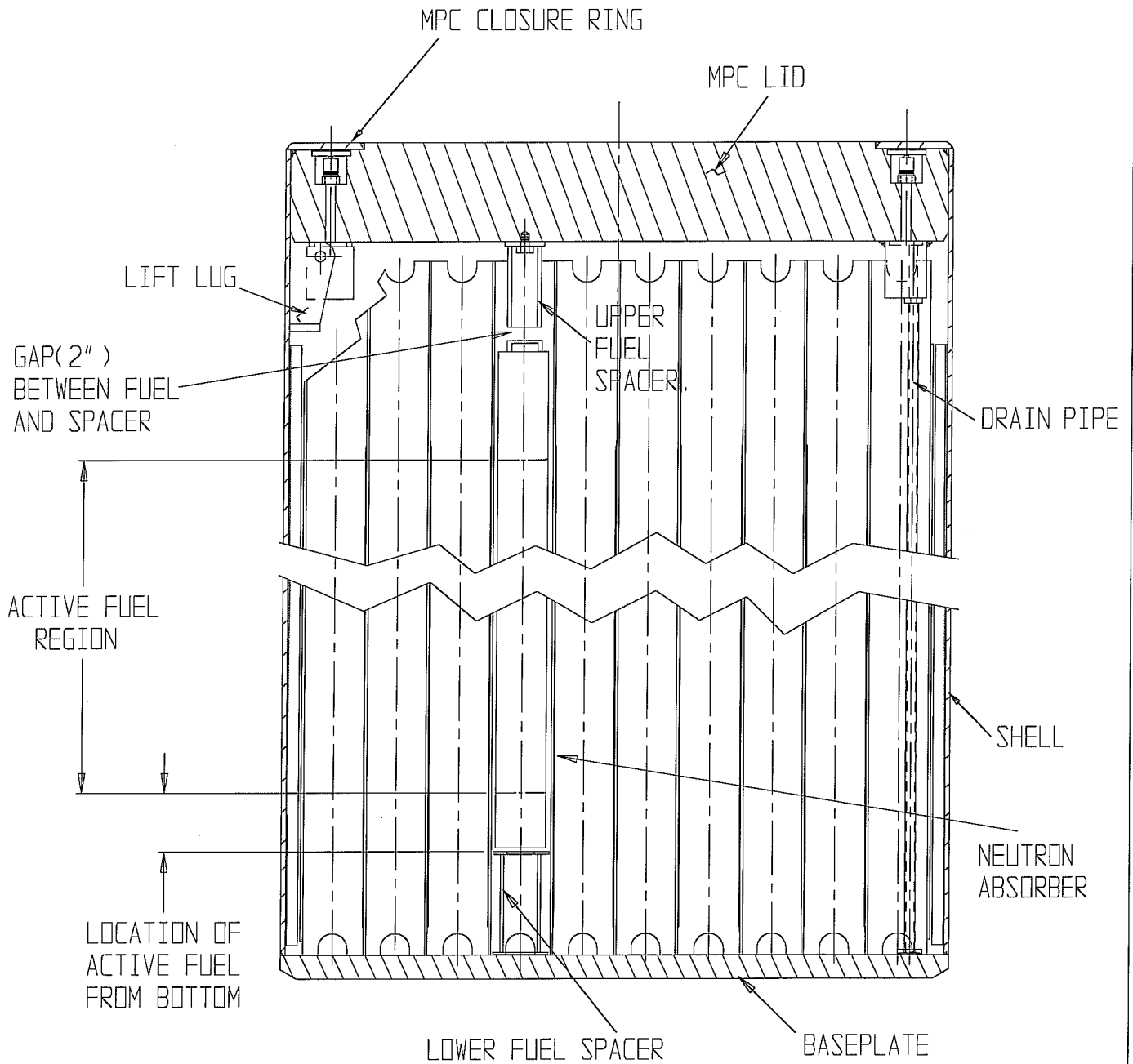


FIGURE 2.1.5; MPC WITH UPPER AND LOWER FUEL SPACERS



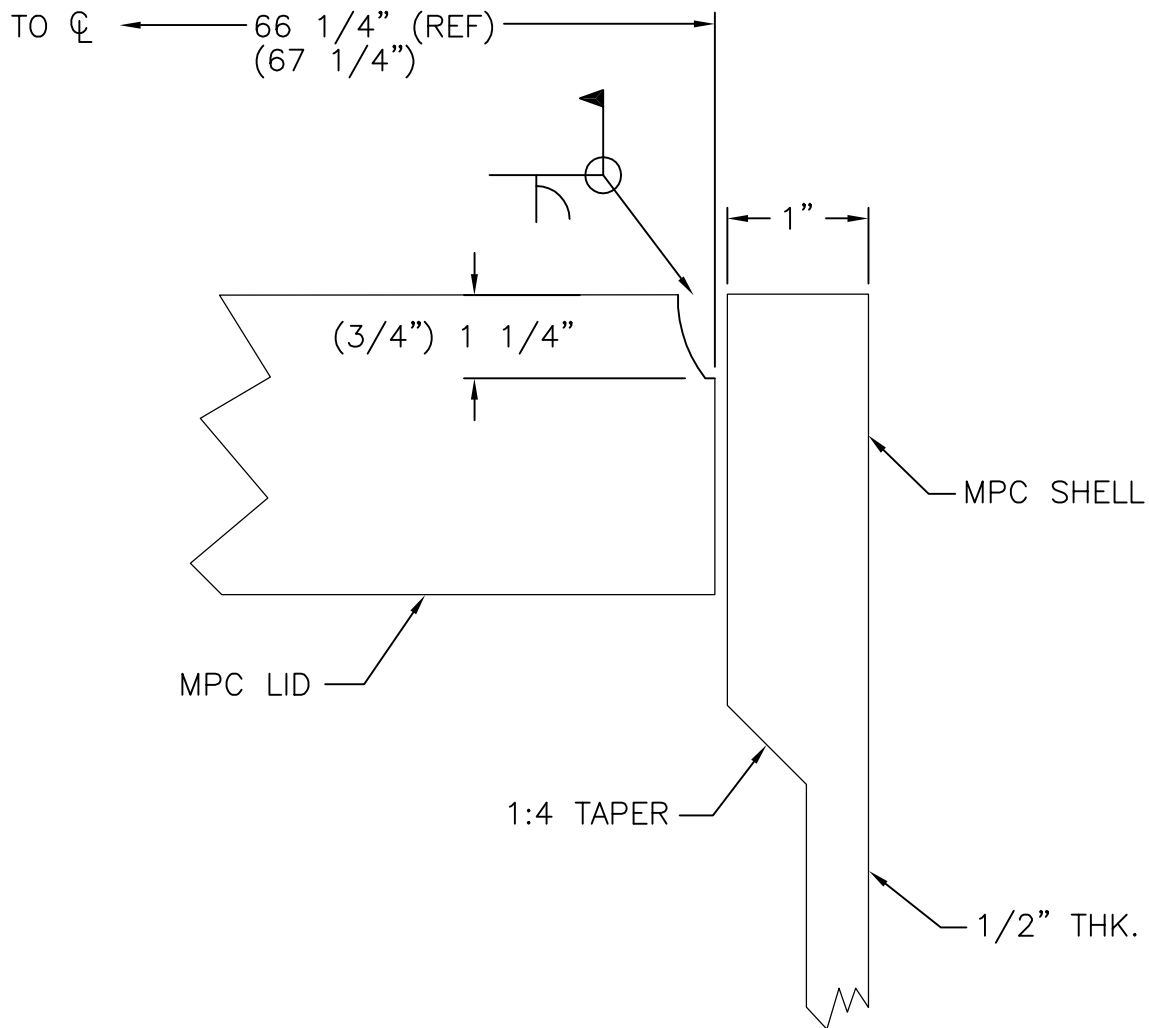
**FIGURE 2.1.6**  
**INTENTIONALLY DELETED**

DELETED

FIGURE 2.1.7; DELETED

DELETED

FIGURE 2.1.8; DELETED



- NOTES: 1. Standard MPC dimensions in parentheses.  
 2. Standard MPC shell thickness is  $\frac{1}{2}''$  along its entire length.  
 3. Figure is not to scale.

Figure 2.1.9; Fuel Debris MPC ( "F" Model)

## 2.2 HI-STORM 100 DESIGN CRITERIA

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

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

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

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

Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Partial Blockage of Air Inlets, Off-Normal Handling of HI-TRAC, *Supplemental Cooling System Power Failure*

Accident Condition: Handling Accident, Tip-Over, Fire, Partial Blockage of MPC Basket Vent Holes, Tornado, Flood, Earthquake, Fuel Rod Rupture, Confinement Boundary Leakage, Explosion, Lightning, Burial Under Debris, 100% Blockage of Air Inlets, Extreme Environmental Temperature, *Supplemental Cooling System Operational Failure*

Short-Term Operations: *This loading condition is defined to accord with ISG-11, Revision 3 guidance [2.0.8]. This includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and on-site handling of a loaded HI-TRAC transfer cask.*

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

## 2.2.1 Normal Condition Design Criteria

### 2.2.1.1 Dead Weight

The HI-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 devices shall meet the requirements of ANSI N14.6<sup>†</sup> [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 radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released in accordance with NUREG-1536.

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<sup>†</sup> 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.

Table 2.2.1 provides the design pressures for the HI-STORM 100 System.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container, it is conservatively assumed that 100% of the fuel rods are ruptured with 100% of the rod fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released for both normal and off-normal conditions. For PWR assemblies stored with non-fuel hardware, it is assumed that 100% of the gasses in the non-fuel hardware (e.g., BPRAs) is also released. This condition is bounded by the pressure calculation for design basis intact fuel with 100% of the fuel rods ruptured in all of the fuel assemblies. It is shown in Chapter 4 that the accident condition design pressure is not exceeded with 100% of the fuel rods ruptured in all of the design basis fuel assemblies. Therefore, rupture of 100% of the fuel rods in the damaged fuel assemblies or fuel debris will not cause the MPC internal pressure to exceed the accident design pressure.

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

The HI-STORM overpack and MPC external pressure is a function of environmental conditions which may produce a pressure loading. The normal and off-normal condition external design pressure is set at ambient standard pressure (1 atmosphere).

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

The HI-TRAC is not capable of retaining internal pressure due to its open design and, therefore, ambient and hydrostatic pressures are the only pressures experienced. Due to the thick steel walls of the HI-TRAC transfer cask, it is evident that the small hydrostatic pressure can be easily withstood; no analysis is required or provided for the HI-TRAC internal pressure. However, the HI-TRAC water jacket does experience internal pressure due to the heat-up of the water contained in the water jacket. Analysis is presented in Chapter 3 that demonstrates that the design pressure in Table 2.2.1 can be withstood by the water jacket and Chapter 4 demonstrates by analysis that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device set at the design pressure is provided, which ensures the pressure will not be exceeded.

#### 2.2.1.4 Environmental Temperatures

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

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

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

#### 2.2.1.5 Design Temperatures

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

Maintaining fuel rod cladding integrity is also a design consideration. The ~~maximum-fuel rod peak~~ *cladding temperature (PCT) limits for the long-term storage and short-term normal operating conditions meet the intent of the guidance in ISG-11, Revision 3 [2.0.8]. For moderate burnup fuel, the previously licensed PCT limit of 570°C (1058°F) may be used [2.0.9] (see also Section 4.5).* ~~calculated by the DCCG (Diffusion Controlled Cavity Growth) methodology outlined in the LLNL report [2.2.14] in accordance with NUREG-1536. However, for conservatism, the PNL methodology outlined in PNL report [2.0.3] produces a lower fuel cladding temperature, which is used to establish the permissible fuel cladding temperature limits, which are used to determine the allowable fuel decay heat load. Maximum fuel rod stainless steel cladding temperature limits recommended in EPRI report [2.2.13] are greater than the long term allowable Zircaloy fuel cladding temperature limits. However, in this FSAR the long term Zircaloy fuel cladding temperature limits are conservatively applied to the stainless steel clad fuel. The short term temperature limits for Zircaloy and stainless steel cladding are taken from references [2.2.15] and [2.2.13], respectively. A detailed description of the maximum fuel rod cladding temperature limits determination is provided in Section 4.3.~~



#### 2.2.1.6 Snow and Ice

The HI-STORM 100 System must be capable of withstanding pressure loads due to snow and ice. ASCE 7-88 (formerly ANSI A58.1) [2.2.2] provides empirical formulas and tables to compute the effective design pressure on the overpack due to the accumulation of snow for the contiguous U.S. and Alaska. Typical calculated values for heated structures such as the HI-STORM 100 System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure loading is set at a level in Table 2.2.8 which bounds the ASCE 7-88 recommendation.

#### 2.2.2 Off-Normal Conditions Design Criteria

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

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

##### 2.2.2.1 Pressure

The HI-STORM 100 System must withstand loads due to off-normal pressure. The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released in accordance with NUREG-1536. ~~For conservatism, the MPC normal internal design pressure bounds both normal and off-normal conditions. Therefore, the normal and off-normal condition MPC internal pressures are set equal for analysis purposes.~~

##### 2.2.2.2 Environmental Temperatures

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

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

STORM 100 storage system which reduces the effect of undulations in instantaneous temperature on the internals of the multi-purpose canister.

### 2.2.2.3 Design Temperatures

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

### 2.2.2.4 Leakage of One Seal

The MPC enclosure vessel is designed to be leak tight under all normal, off-normal, and hypothetical accident conditions of storage. Leakage from the confinement boundary is not credible. ~~HI-STORM 100 System must withstand leakage of one seal in the radioactive material confinement boundary.~~

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

~~Although leakage of one seal is not a credible accident because the MPC confinement boundary does not employ seals, a non-mechanistic leak of the confinement boundary is analyzed as an accident event in Chapter 11.~~

### 2.2.2.5 Partial Blockage of Air Inlets

The HI-STORM 100 System must withstand the partial blockage of the overpack air inlets. This event is conservatively defined as a complete blockage of two (2) of the four air inlets. Because the overpack air inlets and outlets are covered by fine mesh steel screens, located 90° apart, and inspected routinely (or alternatively, exit vent air temperature monitored), it is unlikely that all vents could become blocked by blowing debris, animals, etc. during normal and off-normal operations.

Two of the air inlets are conservatively assumed to be completely blocked to demonstrate the inherent thermal stability of the HI-STORM 100 System.

#### 2.2.2.6 Off-Normal HI-TRAC Handling

During upending and/or downending of the HI-TRAC 100 or HI-TRAC 125 transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation. The HI-TRAC 125D transfer cask design does not include pocket trunnions. Therefore, the entire load is held by the lifting trunnions.

If the lifting device cables begin to “go slack” while upending or downending the HI-TRAC 100 or HI-TRAC 125, the eccentricity of the pocket trunnions would immediately cause the cask to pivot, restoring tension on the cables. Nevertheless, the pocket trunnions are conservatively analyzed to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to demonstrate that the pocket trunnions in the standard HI-TRAC design possess sufficient strength to support the increased load under this off-normal condition.

#### 2.2.3 Environmental Phenomena and Accident Condition Design Criteria

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

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

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

##### 2.2.3.1 Handling Accident

The HI-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 lifting equipment which complies 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 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. For site-specific conditions, which are not encompassed by Table 2.2.9, 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 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. For site-specific conditions, which are not encompassed by Table 2.2.9, 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 analyses in Appendix 3.A* 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 for the HI-TRAC transfer cask and no vertical lift height limit is required to be established. Likewise, while the loaded HI-TRAC is positioned atop the HI-STORM 100 overpack for transfer of the MPC into the overpack (outside the Part 50 facility), the lifting equipment will remain engaged with the lifting trunnions of the HI-TRAC transfer cask or suitable restraints will be provided to secure the HI-TRAC. This ensures that a tip-over or drop from atop the HI-STORM 100 overpack is not a credible accident for the HI-TRAC transfer cask. The design criteria and conditions of use for MPC transfer operations from the HI-TRAC transfer cask to the HI-STORM 100 overpack at a Cask Transfer Facility are specified in ~~the HI-STORM 100 CoC, Appendix B, Section 3.5 and in Subsection 2.3.3.1 of this FSAR.~~

The loaded MPC is lowered into the HI-STORM or HI-STAR overpack or raised from the overpack using the HI-TRAC transfer cask and a MPC lifting system designed in accordance with ANSI N14.6 and having redundant drop protection features. Therefore, the possibility of a loaded MPC falling freely from its highest elevation during the MPC transfer operations into the HI-STORM or HI-STAR overpacks is not credible.

The magnitude of loadings imparted to the HI-STORM 100 System due to drop events is heavily influenced by the compliance characteristics of the impacted surface. Two “pre-approved” concrete pad designs for storing the HI-STORM 100 System are presented in Table 2.2.9. Other ISFSI pad designs may be used provided the designs are reviewed by the Certificate Holder to ensure that impactive and impulsive loads under accident events such as cask drop and non-mechanistic tip-over are less than the design basis limits when analyzed using the methodologies established in this FSAR.

### 2.2.3.2 Tip-Over

The free-standing HI-STORM 100 System is demonstrated by analysis to remain kinematically stable under the design basis environmental phenomena (tornado, earthquake, etc.). However, the HI-STORM 100 Overpack and MPC shall also withstand impacts due to a hypothetical tip-over event. The structural integrity of a loaded HI-STORM 100 System after a tip-over onto a reinforced concrete pad is demonstrated by analysis. The cask tip-over is not postulated as an outcome of any environmental phenomenon or accident condition. The cask tip-over is a non-mechanistic event.

The ISFSI pad for deploying a free-standing HI-STORM overpack must possess sufficient structural stiffness to meet the strength limits set forth in the ACI Code selected by the ISFSI owner. At the same time, the pad must be sufficiently compliant such that the maximum deceleration under a tip-over event is below the limit set forth in Table 3.1.2 of this FSAR.

During original licensing for the HI-STORM 100 System, a single set of ISFSI pad and subgrade design parameters (now labeled Set A) was established. Experience has shown that achieving a maximum concrete compressive strength (at 28 days) of 4,200 psi can be difficult. Therefore, a second set of ISFSI pad and subgrade design parameters (labeled Set B) has been developed. The Set B ISFSI parameters include a thinner concrete pad and less stiff subgrade, which allow for a higher concrete compressive strength. Cask deceleration values for all design basis drop and tipover events with the HI-STORM 100 and HI-STORM 100S overpacks have been verified to be less than or equal to the design limit of 45 g’s for both sets of ISFSI pad parameters.

The original set and the new set (Set B) of acceptable ISFSI pad and subgrade design parameters are specified in Table 2.2.9. Users may design their ISFSI pads and subgrade in compliance with either parameter Set A or Set B. Alternatively, users may design their site-specific ISFSI pads and subgrade using any combination of design parameters resulting in a structurally competent pad that meets the provisions of ACI-318 and also limits the deceleration of the cask to less than or equal to 45 g’s for the design basis drop and tip-over events for the HI-STORM 100 and HI-STORM 100S overpacks. The structural analyses for site-specific ISFSI pad design shall be performed using methodologies consistent with those described in this FSAR, as applicable.

If the HI-STORM 100 cask is deployed in an anchored configuration (HI-STORM 100A), then tip-over of the cask is structurally precluded along with the requirement of target compliance, which warrants setting specific limits on the concrete compressive strength and subgrade Young's Modulus. Rather, at the so-called high seismic sites (ZPAs greater than the limit set forth in the CoC for free standing casks), the ISFSI pad must be sufficiently rigid to hold the anchor studs and maintain the integrity of the fastening mechanism embedded in the pad during the postulated seismic event. The ISFSI pad must be designed to minimize a physical uplift during extreme environmental event (viz., tornado missile, DBE, etc.). The requirements on the ISFSI pad to render the cask anchoring function under long-term storage are provided in Section 2.0.4.

#### 2.2.3.3 Fire

The possibility of a fire accident near an ISFSI site is considered to be extremely remote due to the absence of significant combustible materials. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM 100 overpack or HI-TRAC transfer cask while it is being moved to the ISFSI.

The HI-STORM 100 System must withstand temperatures due to a fire event. The HI-STORM overpack and HI-TRAC transfer cask fire accidents for storage are conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible transporter fuel. The HI-STORM overpack and HI-TRAC transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Table 2.2.8 provides the fire durations for the HI-STORM 100 overpack and HI-TRAC transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475° F in accordance with 10CFR71.73.

The accident condition design temperatures for the HI-STORM 100 System, and the fuel rod cladding limits are specified in Table 2.2.3. The specified fuel cladding temperature limits are based on the ~~short-term~~ temperature limits specified in *ISG-11, Rev. 3 [2.0.9]*, reports [2.2.13 and 2.2.15].

#### 2.2.3.4 Partial Blockage of MPC Basket Vent Holes

The HI-STORM 100 System is designed to withstand reduction of flow area due to partial blockage of the MPC basket vent holes. As the MPC basket vent holes are internal to the confinement barrier, the only events that could partially block the vents are fuel cladding failure and debris associated with this failure, or the collection of crud at the base of the stored SNF assembly. The HI-STORM 100 System maintains the SNF in an inert environment with fuel rod cladding temperatures below accepted values (Table 2.2.3). Therefore, there is no credible mechanism for gross fuel cladding degradation during storage in the HI-STORM 100. For the storage of damaged BWR fuel assemblies or fuel debris, the assemblies and fuel debris will be placed in damaged fuel containers prior to placement in the MPC. The damaged fuel container is equipped with fine mesh screens which ensure that the damaged fuel and fuel debris will not escape to block the MPC basket vent holes. In addition, each MPC will be loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities reported in an Empire State Electric Energy Research Corporation Report [2.2.6], a layer of crud of conservative depth is assumed to partially block the MPC basket vent holes. The crud depths for the different MPCs are listed in Table 2.2.8.

### 2.2.3.5 Tornado

The HI-STORM 100 System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STORM 100 System are consistent with NRC Regulatory Guide 1.76 [2.2.7], ANSI 57.9 [2.2.8], and ASCE 7-88 [2.2.2]. Table 2.2.4 provides the wind speeds and pressure drops which the HI-STORM 100 overpack must withstand while maintaining kinematic stability. The pressure drop is bounded by the accident condition MPC external design pressure.

The kinematic stability of the HI-STORM overpack, and continued integrity of the MPC confinement boundary, while within the storage overpack or HI-TRAC transfer cask, must be demonstrated under impact from tornado-generated missiles in conjunction with the wind loadings. Standard Review Plan (SRP) 3.5.1.4 of NUREG-0800 [2.2.9] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile that deforms on impact (large missile); a rigid missile to test penetration resistance (penetrant missile); and a small rigid missile of a size sufficient to pass through any openings in the protective barriers (micro-missile). SRP 3.5.1.4 suggests an automobile for a large missile, a rigid solid steel cylinder for the penetrant missile, and a solid sphere for the small rigid missile, all impacting at 35% of the maximum horizontal wind speed of the design basis tornado. Table 2.2.5 provides the missile data used in the analysis, which is based on the above SRP guidelines. The effects of a large tornado missile are considered to bound the effects of a light general aviation airplane crashing on an ISFSI facility.

During horizontal handling of the loaded HI-TRAC transfer cask outside the Part 50 facility, tornado missile protection shall be provided to prevent tornado missiles from impacting either end of the HI-TRAC. The tornado missile protection shall be designed such that the large tornado missile cannot impact the bottom or top of the loaded HI-TRAC, while in the horizontal position. Also, the missile protection for the top of the HI-TRAC shall be designed to preclude the penetrant missile and micro-missile from passing through the penetration in the HI-TRAC top lid, while in the horizontal position. With the tornado missile protection in place, the impacting of a large tornado missile on either end of the loaded HI-TRAC or the penetrant missile or micro-missile entering the penetration of the top lid is not credible. Therefore, no analyses of these impacts are provided.

### 2.2.3.6 Flood

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

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

The flood water is assumed to be nonstagnant. The maximum allowable flood water velocity is determined by calculating the equivalent pressure loading required to slide or tip over the HI-

STORM 100 System. The design basis flood water velocity is stated in Table 2.2.8. Site-specific safety reviews by the licensee must confirm that flood parameters do not exceed the flood depth, slide, or tip-over forces.

If the flood water depth exceeds the elevation of the top of the HI-STORM overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the HI-STORM 100 Overpack. Blockage of the air inlets is addressed in Subsection 2.2.3.12.

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

Analysis for each site for such transient hydrological loadings must be made for that site. It is expected that the plant licensee will perform this evaluation under the provisions of 10CFR72.212.

#### 2.2.3.7 Seismic Design Loadings

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

For anchored casks, the limit on zero period accelerations is set by the structural capacity of the sector lugs and anchoring studs. Table 2.2.8 provides the limits for HI-STORM 100A for the maximum vector sum of two horizontal earthquake peak ZPA's along with the coincident limit on the vertical ZPA.

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



#### 2.2.3.8 100% Fuel Rod Rupture

The HI-STORM 100 System must withstand loads due to 100% fuel rod rupture. For conservatism, 100 percent of the fuel rods are assumed to rupture with 100 percent of the fill gas and 30% of the significant radioactive gases (e.g., H<sup>3</sup>, Kr, and Xe) released in accordance with NUREG-1536. All of the fill gas contained in non-fuel hardware, such as Burnable Poison Rod Assemblies (BPRAs) is also assumed to be released in analyzing this event.

#### 2.2.3.9 Confinement Boundary Leakage

No credible scenario has been identified that would cause failure of the confinement system. *Section 7.1 provides a discussion as to why leakage of any magnitude from the MPC is not credible, based on the materials and methods of fabrication and inspection.* ~~To demonstrate the overall safety of the HI-STORM 100 System, the largest test leakage rate for the confinement boundary plus 50% for conservatism is assumed as the maximum credible confinement boundary leakage rate and 100 percent of the fuel rods are assumed to have failed. Under this accident condition, doses to an individual located at the boundary of the controlled area are calculated.~~

#### 2.2.3.10 Explosion

The HI-STORM 100 System must withstand loads due to an explosion. The accident condition MPC external pressure and overpack pressure differential specified in Table 2.2.1 bounds all credible external explosion events. There are no credible internal explosive events since all materials are compatible with the various operating environments, as discussed in Section 3.4.1, *or appropriate preventive measures are taken to preclude internal explosive events (see Section 1.2.1.3.1.1).* The MPC is composed of stainless steel, ~~Boral~~ *neutron absorber material*, and, *prior to CoC Amendment 2, possibly optional aluminum alloy 1100 heat conduction elements.*, ~~all of which have a long proven history of use in fuel pools at nuclear power plants.~~ For these materials, *and considering the protective measures taken during loading and unloading operations* there is no credible cause for an internal explosive event.

#### 2.2.3.11 Lightning

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

#### 2.2.3.12 Burial Under Debris

The HI-STORM 100 System must withstand burial under debris. Such debris may result from floods, wind storms, or mud slides. Mud slides, blowing debris from a tornado, or debris in flood water may result in duct blockage, which is addressed in Subsection 2.2.3.13. The thermal effects of burial under debris on the HI-STORM 100 System is evaluated in Chapter 11. Siting of the ISFSI pad shall ensure that the storage location is not located near shifting soil. Burial under debris is a highly unlikely accident, but is analyzed in this FSAR.

#### 2.2.3.13 100% Blockage of Air Inlets

For conservatism, this accident is defined as a complete blockage of all four bottom air inlets. Such a blockage may be postulated to occur during accident events such as a flood or tornado with blowing debris. The HI-STORM 100 System must withstand the temperature rise as a result of 100% blockage of the air inlets and outlets. The fuel cladding temperature must be shown to remain below the ~~short term~~ *off-normal/accident* temperature limit specified in Table 2.2.3.

#### 2.2.3.14 Extreme Environmental Temperature

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

#### 2.2.3.15 Bounding Hydraulic, Wind, and Missile Loads for HI-STORM 100A

In the anchored configuration, the HI-STORM 100A System is clearly capable of withstanding much greater lateral loads than a free-standing overpack. Coastal sites in many areas of the world, particularly the land mass around the Pacific Ocean, may be subject to severe fluid inertial loads. Several publications [2.2.10, 2.2.11] explain and quantify the nature and source of such environmental hazards.

It is recognized that a lateral fluid load may also be accompanied by an impact force from a fluid borne missile (debris). Rather than setting specific limits for these loads on an individual basis, a limit on the static overturning base moment on the anchorage is set. This bounding overturning moment is given in Table 2.2.8 and is set at a level that ensures that structural safety margins on the sector lugs and on the anchor studs are essentially equal to the structural safety margins of the same components under the combined effect of the net horizontal and vertical seismic load limits in Table 2.2.8. The ISFSI owner bears the responsibility to establish that the lateral hydraulic, wind, and missile loads at his ISFSI site do not yield net overturning moments, when acting separately or together, that exceed the limit value in Table 2.2.8. If loadings are increased above those values for free-standing casks, their potential effect on the other portions of the cask system must be considered.

#### 2.2.4 Applicability of Governing Documents

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition, with Addenda through 1997 [2.2.1], is the governing code for the structural design of the MPC, the metal structure of the HI-STORM 100 overpack, and the HI-TRAC transfer cask, *except for Sections V and IX. The latest effective editions of ASME Section V and IX may be used, provided a written reconciliation of the later edition against the 1995 Edition, including addenda, is performed by the certificate holder.* The MPC enclosure vessel and fuel basket are designed in accordance with Section III, Subsections NB Class 1 and NG Class 1, respectively. The metal structure of the overpack and the HI-TRAC transfer

cask are designed in accordance with Section III, Subsection NF Class 3. The ASME Code is applied to each component consistent with the function of the component.

ACI 349 is the governing code for the plain concrete in the HI-STORM 100 overpack. ACI 318-95 is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited during structural analysis. Appendix 1.D provides the sections of ACI 349 and ACI 318-95 applicable to the plain concrete.

Table 2.2.6 provides a summary of each structure, system and component (SSC) of the HI-STORM 100 System that is identified as important to safety, along with its function and governing Code. Some components perform multiple functions and in those cases, the most restrictive Code is applied. In accordance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components", and according to importance to safety, components of the HI-STORM 100 System are classified as A, B, C, or NITS (not important to safety) in Table 2.2.6. Section 13.1 provides the criteria used to classify each item. The classification of necessary auxiliary equipment is provided in Table 8.1.6.

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

Table 2.2.15 lists the ~~exceptions~~ *alternatives* to the ASME Code for the HI-STORM 100 System and the justification for those ~~exceptions~~ *alternatives*.

The MPC *enclosure vessel and certain fuel basket designs* utilized in the HI-STORM 100 System ~~are~~ identical to the MPC *components* described in the SARs for the HI-STAR 100 System for storage (Docket 72-1008) and transport (Docket 71-9261). To avoid unnecessary repetition of the large numbers of stress analyses, ~~attention is directed in this document~~ *refers to those SARs, as applicable, if to establish that* the MPC loadings for storage in the HI-STORM 100 System do not exceed those computed in the ~~referenced applications~~ *HI-STAR documents*. Many of the loadings in the HI-STAR applications envelope the HI-STORM loadings on the MPC, and, therefore, a complete re-analysis of the MPC is not provided in the FSAR. *Certain individual MPC analyses may have been required to license a particular MPC fuel basket design for HI-STORM that was not previously licensed for HI-STAR. These unique analyses are summarized in the appropriate location in this FSAR.*

Table 2.2.16 provides a summary comparison between the loading elements. Table 2.2.16 shows that most of the loadings remain unchanged and several are less than the HI-STAR loading conditions. In addition to the magnitude of the loadings experienced by the MPC, the application of the loading must also be considered. Therefore, it is evident from Table 2.2.16 that the MPC stress limits can be ascertained to be qualified a priori if the HI-STAR analyses and the thermal loadings under HI-STORM storage are not more severe compared to previously analyzed HI-STAR conditions. In the analysis of each of the normal, off-normal, and accident conditions, the effect on the MPC is evaluated and compared to the corresponding condition analyzed in the HI-STAR 100 System SARs [2.2.4 and 2.2.5]. If the HI-STORM loading is greater than the HI-STAR loading or the loading is applied differently, the analysis of its effect on the MPC is evaluated in Chapter 3.

### 2.2.5 Service Limits

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

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

- a. Level A Service Limits: Level A Service Limits are used to establish allowables for normal condition load combinations.
- b. Level B Service Limits: Level B Service Limits are used to establish allowables for off-normal condition load combinations.
- c. Level C Service Limits: Level C Service Limits are not used.
- d. Level D Service Limits: Level D Service Limits are used to establish allowables for accident condition load combinations.

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

The MPC confinement boundary is required to meet Section III, Class 1, Subsection NB stress intensity limits. Table 2.2.10 lists the stress intensity limits for the Levels A, B, C, and D service limits for Class 1 structures extracted from the ASME Code (1995 Edition). The limits for the MPC fuel basket, required to meet the stress intensity limits of Subsection NG of the ASME Code, are listed in Table 2.2.11. Table 2.2.12 lists allowable stress limits for the steel structure of the HI-STORM overpack and HI-TRAC which are analyzed to meet the stress limits of Subsection NF, Class 3. Only service levels A, B, and D requirements, normal, off-normal, and accident conditions, are applicable.

### 2.2.6 Loads

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

The loadings listed in Table 2.2.13 fall into two broad categories; namely, (i) those that primarily affect kinematic stability, and (ii) those that produce significant stresses. The loadings in the former

category are principally applicable to the overpack. Tornado wind ( $W'$ ), earthquake (E), and tornado-borne missile (M) are essentially loadings which can destabilize a cask. Analyses reported in Chapter 3 show that the HI-STORM 100 overpack structure will remain kinematically stable under these loadings. Additionally, for the missile impact case (M), analyses that demonstrate that the overpack structure remains unbreached by the postulated missiles are provided in Chapter 3.

Loadings in the second category produce global stresses that must be shown to comply with the stress intensity or stress limits, as applicable. The relevant loading combinations for the fuel basket, the MPC, the HI-TRAC and the HI-STORM overpack are different because of differences in their function. For example, the fuel basket does not experience a pressure loading because it is not a pressure vessel. The specific load combination for each component is specified in Subsection 2.2.7.

### 2.2.7 Load Combinations

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

For example, the values established for internal and external pressures ( $P_i$  and  $P_o$ ) are defined such that they bound other surface-intensive loads, namely snow (S), tornado wind ( $W'$ ), flood (F), and explosion ( $E^*$ ). Thus, evaluation of pressure in a load combination established for a given storage condition enables many individual load effects to be included in a single load combination.

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

Table 2.2.1

DESIGN PRESSURES

<b>Pressure Location</b>	<b>Condition</b>	<b>Pressure (psig)</b>
MPC Internal Pressure	Normal	100
	Off-Normal	<del>100</del> 110
	Accident	200
MPC External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	60
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	10 (differential pressure for 1 second maximum) or 5 (differential pressure steady state)
HI-TRAC Water Jacket	Normal	60
	Off-normal	60
	Accident	N/A (Under accident conditions, the water jacket is assumed to have lost all water thru the pressure relief valves)

Table 2.2.2

ENVIRONMENTAL TEMPERATURES

Condition	Temperature (°F)	Comments
HI-STORM 100 Overpack		
Normal Ambient (Bounding Annual Average)	80	
Normal Soil Temperature (Bounding Annual Average)	77	
Off-Normal Ambient (3-Day Average)	-40 and 100	-40°F with no insolation  100°F with insolation
Extreme Accident Level Ambient (3-Day Average)	125	125°F with insolation starting at steady-state off-normal high environment temperature
HI-TRAC Transfer Cask		
Normal (Bounding Annual Average)	100	
Off-Normal (3-Day Average)	0 and 100	0° F with no insolation  100° F with insolation

Note:

1. Handling operations with the loaded HI-STORM 100 overpack and HI-TRAC transfer cask are limited to working area ambient temperatures greater than or equal to 0°F as specified in Subsection 2.2.1.2. ~~and the Design Features section of Appendix B to the CoC.~~



Table 2.2.3

## DESIGN TEMPERATURES

<b>HI-STORM 100 Component</b>	<b>Long Term, Normal Condition Design Temperature- Limits (° F)</b>	<b>Short Term Operations, Off-Normal, and Accident Condition Temperature- Limits<sup>†</sup> (° F)</b>
MPC shell	450	775
MPC basket	725	950
MPC Boral Neutron Absorber	800	950
MPC lid	550	775
MPC closure ring	400	775
MPC baseplate	400	775
MPC Heat Conduction Elements	725	950
HI-TRAC inner shell	400	600
HI-TRAC pool lid/transfer lid	350	700
HI-TRAC top lid	400	700
HI-TRAC top flange	400	700
HI-TRAC pool lid seals	350	N/A
HI-TRAC bottom lid bolts	350	700
HI-TRAC bottom flange	350	700
HI-TRAC top lid neutron shielding	300	300/350
HI-TRAC radial neutron shield	307	N/A
HI-TRAC radial lead gamma shield	350	600
Remainder of HI-TRAC	350	700
<i>Fuel Cladding</i>	752	752 or 1058 (Short Term Operations) <sup>††</sup>  1058 (Off-normal and Accident Conditions)
Zircaloy fuel cladding (5-year cooled) <sup>†</sup>	-691(PWR) -740(BWR)	1058
Zircaloy fuel cladding (6-year cooled) <sup>†</sup>	-676(PWR) -712(BWR)	1058
Zircaloy fuel cladding (7-year cooled) <sup>†</sup>	-635(PWR) -669(BWR)	1058

<sup>†</sup> For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the ISFSI fire event, the maximum temperature limit for ASME Section 1 equipment is appropriate (850°F in Code Table 1A).

<sup>††</sup> Normal short term operations includes MPC drying and onsite transport per Reference [2.0.8]. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel as discussed in Reference [2.0.9]. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F. See also Section 4.3.

Table 2.2.3 (continued)  
DESIGN TEMPERATURES

<b>HI-STORM 100 Component</b>	<b>Long Term, Normal Condition Design Temperature: Limits (° F)</b>	<b>Short Term Operations, Off-Normal, and Accident Condition Temperature: Limits<sup>†</sup> (° F)</b>
Zircaloy fuel cladding (10-year cooled) <sup>1</sup>	-625(PWR) -658(BWR)	1058
Zircaloy fuel cladding (15-year cooled) <sup>1</sup>	-614(PWR) -646(BWR)	1058
Zircaloy fuel cladding (5-year cooled) <sup>2</sup>	679 (PWR) 740 (BWR)	1058
Zircaloy fuel cladding (6-year cooled) <sup>2</sup>	660 (PWR) 712 (BWR)	1058
Zircaloy fuel cladding (7-year cooled) <sup>2</sup>	635 (PWR) 669 (BWR)	1058
Zircaloy fuel cladding (10-year cooled) <sup>2</sup>	621 (PWR) 658 (BWR)	1058
Zircaloy fuel cladding (15-year cooled) <sup>2</sup>	611 (PWR) 646 (BWR)	1058
Overpack outer shell	350	600
Overpack concrete	300/200	350
Overpack inner shell	350	400
Overpack Lid Top and Bottom Plate	350 450	550
Remainder of overpack steel structure	350	400

NOTES: ——— 1. Moderate Burnup Fuel  
 ——— 2. High Burnup Fuel (see Table 4.A.2)

Table 2.2.4

TORNADO CHARACTERISTICS

Condition	Value
Rotational wind speed (mph)	290
Translational speed (mph)	70
Maximum wind speed (mph)	360
Pressure drop (psi)	3.0

Table 2.2.5

TORNADO-GENERATED MISSILES

<b>Missile Description</b>	<b>Mass (kg)</b>	<b>Velocity (mph)</b>
Automobile	1800	126
Rigid solid steel cylinder (8 in. diameter)	125	126
Solid sphere (1 in. diameter)	0.22	126

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
MPC<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Confinement	Shell	A	ASME Section III; Subsection NB	Alloy X <sup>(5)</sup>	See Appendix 1.A	NA	NA
Confinement	Baseplate	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Lid	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Closure Ring	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Port Cover Plates	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Basket Cell Plates	A	ASME Section III; Subsection NG; <i>core support structures (NG-1121)</i>	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	<del>Boral</del> Neutron Absorber	A	Non-code	NA	NA	NA	Aluminum/SS
Shielding	Drain and Vent Shield Block	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Shielding	Plugs for Drilled Holes	NITS	Non-code	SA 193B8 (or equivalent)	See Appendix 1.A	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) For details on Alloy X material, see Appendix 1.A.
  - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength  $\geq 75$  ksi, yield strength  $\geq 30$  ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
MPC<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Upper Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Sheathing	A	Non-code	Alloy X	See Appendix 1.A	Aluminum/SS	NA
Structural Integrity	Shims	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Angled Plates)	A	ASME Section III; Subsection NG; <i>internal structures (NG-1122)</i>	Alloy X	See Appendix 1.A	NA	NA
Structural Form	Basket Supports (Flat Plates)	NITS	Non-Code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug	C	NUREG-0612	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug Baseplate	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Bolt	NITS	Non-code	A193-B8 (or equiv.)	Per ASME Section II	NA	NA
Structural Integrity	Upper Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Stainless Steel. See Note 6	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA

- 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 Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
  - 5) For details on Alloy X material, see Appendix 1.A.
  - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength  $\geq 75$  ksi, yield strength  $\geq 30$  ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
MPC<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Vent Shield Block Spacer	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Vent and Drain Tube	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	NA
Operations	Vent & Drain Cap	C	Non-code	S/S	Per ASME Section II	NA	NA
Operations	Vent & Drain Cap Seal Washer	NITS	Non-code	Aluminum	NA	NA	Aluminum/SS
Operations	Vent & Drain Cap Seal Washer Bolt	NITS	Non-code	Aluminum	NA	NA	NA
Operations	Reducer	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Drain Line	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Damaged Fuel Container	C	ASME Section III; Subsection NG	Primarily 304 S/S, except for locking spring, which may be any type of SS	See Appendix 1.A	NA	NA
Operations	Drain Line Guide Tube	NITS	Non-code	S/S	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
  - 5) For details on Alloy X material, see Appendix 1.A.
  - 6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength  $\geq 75$  ksi, yield strength  $\geq 30$  ksi and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
OVERPACK<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Shielding	Radial Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Block Ring (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Shielding	Lid Shield Ring (100S) and Shield Block Shell (100S)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Shielding	Shield Block Shell (100)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Shielding	Pedestal Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Lid Shield	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Shield Shell (eliminated from design 6/01)	B	See Note 6	SA516-70	See Table 3.3.2	NA	NA
Shielding	Shield Block	B	ACI 349, App. 1-D	Concrete	See Table 1.D.1	NA	NA
Shielding	Gamma Shield Cross Plates & Tabs	C	Non-code	SA240-304	NA	NA	NA
Structural Integrity	Baseplate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.3	See Note 5	NA
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Concrete Form	Pedestal Shell	B	See Note 6	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 ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
  - 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  
OVERPACK<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Concrete Form	Pedestal Plate (100) Pedestal Baseplate (100S)	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Table 3.3.2	NA
Structural Integrity	Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inlet Vent Vertical & Horizontal Plates	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Thermal	Exit Vent Horizontal Plate (100)	B	See Note 6	SA516-70	See Table 3.3.2	See Note 5	NA
Thermal	Exit Vent Vertical/Side Plate	B	See Note 6	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Plate, including shear ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Top Plate, including shear ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial 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 ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
  - 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  
OVERPACK<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Lid Stud & Nut	B	ASME Section II	SA564-630 or SA 193-B7 (stud) SA 194-2H (nut)	See Table 3.3.4	Threads to have cadmium coating (or similar lubricant for corrosion protection)	NA
Structural Integrity	100S Lid Washer	B	Non-Code	SA240-304	Per ASME Section II	NA	NA
Structural Integrity	Bolt Anchor Block	B	ASME Section III; Subsection NF ANSI N14.6	SA350-LF3 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 (not applicable to SA240-304)	NA
Structural Integrity	Channel Mounts	B	ASME Section III; Subsection NF	A36 or equivalent	Per ASME Section II	See Note 5	NA
Structural Integrity	Pedestal Platform	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	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

- 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 ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
  - 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  
OVERPACK <sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Operations	Temperature Element & Associated Temperature Monitoring Equipment	B	Non-code	NA	NA	NA	NA
Operations	Screen	NITS	Non-code	Mesh Wire	NA	NA	NA
Operations	Paint	NITS	Non-code	Thermaline 450 or equivalent	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All exposed steel surfaces (except threaded holes) to be painted with Thermaline 450 or equivalent.
  - 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<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Shielding	Radial Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Pool Lid Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Top Lid Shielding	B	Non-code	Holtite	NA	NA	NA
Shielding	Plugs for Lifting Holes	NITS	Non-code	C/S	NA	NA	
Structural Integrity	Outer Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Inner Shell	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Radial Ribs	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket Enclosure Shell Panels (HI-TRAC 100 and 125)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket Enclosure Shell Panels (HI-TRAC 125D)	B	ASME Section III; Subsection NF	SA516-70 or SA515-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Water Jacket End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Flange	B	ASME Section III; Subsection NF	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lower Water Jacket Shell	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 Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
HI-TRAC TRANSFER CASK <sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Structural Integrity	Pool Lid Outer Ring	B	ASME Section III; Subsection NF	SA516-70 or SA 203E or SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Pool Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Outer Ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Inner Ring	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Fill Port Plugs	C	ASME Section III; Subsection NF	Carbon Steel	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 Chapter 13 *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
HI-TRAC TRANSFER CASK<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	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 ANSI N14.6	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lifting Trunnion	A	ANSI N14.6	SB637 (N07718)	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 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
Structural Integrity	Pool Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Top Lid Lifting Block	C	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Bottom Flange Gussets (HI-TRAC 125D only)	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	NA	NA
Operations	Top Lid Stud or bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
HI-TRAC TRANSFER CASK <sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Operations	Top Lid Nut	B	ASME Section III; Subsection NF	SA194-2H	NA	NA	NA
Operations	Pool Lid Gasket	NITS	Non-code	Elastomer	NA	NA	NA
Operations	Lifting Trunnion End Cap (HI-TRAC 100 and HI-TRAC 125 only)	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	End Cap Bolts (HI-TRAC 100 and HI-TRAC 125 only)	NITS	Non-code	SA193-B7	See Table 3.3.4	NA	NA
Operations	Drain Pipes	NITS	Non-code	SA106	NA	NA	NA
Operations	Drain Bolt	NITS	Non-code	SA193-B7	See Table 3.3.4	NA	NA
Operations	Couplings, Valves and Vent Plug	NITS	Non-code	Commercial	NA	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in Chapter 13 the Holtec QA Program. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Inside surface of transfer cask to be painted with Thermaline 450.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
HI-TRAC TRANSFER LID (HI-TRAC 100 and HI-TRAC 125 ONLY)<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Shielding	Side Lead Shield	B	Non-code	Lead	NA	NA	NA
Shielding	Door Lead Shield	B	Non-code	Lead	NA	NA	
Shielding	Door Shielding	B	Non-code	Holtite	NA	NA	NA
Structural Integrity	Lid Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lid Intermediate Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lead Cover Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lead Cover Side Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Top Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Middle Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Bottom Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Wheel Housing	B	ASME Section III; Subsection NF	SA516-70 (SA350-LF3)	See Table 3.3.2 (Table 3.3.3)	See Note 5	NA
Structural Integrity	Door Interface 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 ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Top surface of doors to be painted with Thermaline 450.



TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM  
HI-TRAC TRANSFER LID (HI-TRAC 100 and HI-TRAC 125 ONLY)<sup>(1,2)</sup>

Primary Function	Component <sup>(3)</sup>	Safety Class <sup>(4)</sup>	Codes/Standards ( as applicable to component)	Material	Strength ( ksi)	Special Surface Finish/Coating	Contact Matl. ( if dissimilar)
Structural Integrity	Door Side Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Wheel Shaft	C	ASME Section III; Subsection NF	SA 193-B7	36 ( yield)	See Note 5	NA
Structural Integrity	Lid Housing Stiffener	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Door Lock Bolt	B	ASME Section III; Subsection NB	SA193-B7	See Table 3.3.4	NA	NA
Structural Integrity	Door End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Structural Integrity	Lifting Lug and Pad	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Wheel Track	C	ASME Section III; Subsection NF	SA-36	36 ( yield)	See Note 5	NA
Operations	Door Handle	NITS	Non-code	C/S or S/S	NA	See Note 5	NA
Operations	Door Wheels	NITS	Non-code	Forged Steel	NA	NA	NA
Operations	Door Stop Block	C	Non-code	SA516-70	See Table 3.3.2	See Note 5	NA
Operations	Door Stop Block Bolt	C	Non-code	SA193-B7	See Table 3.3.4	NA	NA

- Notes:
- 1) There are no known residuals on finished component surfaces
  - 2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
  - 3) Component nomenclature taken from Bill of Materials in Chapter 1.
  - 4) A, B, and C denote important to safety classifications as described in ~~Chapter 13~~ *the Holtec QA Program*. NITS stands for Not Important to Safety.
  - 5) All external surfaces to be painted with Carboline 890. Top surface of doors to be painted with Thermaline 450.

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; <i>core support structures (NG-1121)</i>	Section III, Subsection NG, NG-3300 and NG-3200; <i>core support structures (NG-1121)</i>	Section III, Subsection NG, NG-4000; <i>core support structures (NG-1121)</i>	Section III, Subsection NG, NG-5000 and Section V; <i>core support structures (NG-1121)</i>
HI-TRAC Lifting Trunnions	Section II, Section III, Subsection NF, NF-2000	ANSI N14.6	Section III, Subsection NF, NF-4000	See Chapter 9
MPC basket supports ( <i>Angled Plates</i> )	Section II, Section III, Subsection NG, NG-2000; <i>internal structures (NG-1122)</i>	Section III, Subsection NG, NG-3300 and NG-3200; <i>internal structures (NG-1122)</i>	Section III, Subsection NG, NG-4000; <i>internal structures (NG-1122)</i>	Section III, Subsection NG, NG-5000 and Section V; <i>internal structures (NG-1122)</i>
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 318-95 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.

Table 2.2.8

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

Item	Condition	Value
Snow Pressure Loading (lb./ft <sup>2</sup> )	Normal	100
Constriction of MPC Basket Vent Opening By Crud Settling (Depth of Crud, in.)	Accident	0.85 (MPC-68) 0.36 (MPC-24 and MPC-32)
Cask Environment During the Postulated Fire Event (Deg. F)	Accident	1475
HI-STORM Overpack Fire Duration (seconds)	Accident	217
HI-TRAC Transfer Cask Fire Duration (minutes)	Accident	4.8
Maximum submergence depth due to flood (ft)	Accident	125
Flood water velocity (ft/s)	Accident	15
Interaction Relation for Horizontal & Vertical <del>acceleration</del> <del>ZPA (Zero Period Acceleration)</del> for HI-STORM	Accident	$G_H + 0.53G_V = 0.53^{\dagger\dagger}$ (HI-STORM 100 and 100S)  $G_H = 2.12; G_V = 1.5$ (HI-STORM 100A)
Net Overturning Moment at base of HI-STORM 100A (ft-lb)	Accident	$18.7 \times 10^6$
HI-STORM 100 Overpack Vertical Lift Height Limit (in.)	Accident	$11^{\dagger\dagger\dagger}$ (HI-STORM 100 and 100S), OR By Users (HI-STORM 100A)
HI-TRAC Transfer Cask Horizontal Lift Height Limit (in.)	Accident	$42^{\dagger\dagger\dagger}$

<sup>††</sup> See Subsection 3.4.7.1 for definition of  $G_H$  and  $G_V$ . The coefficient of ~~friction~~ ~~0.53~~ may be increased *above 0.53* based on testing described in Subsection 3.4.7.1

<sup>†††</sup> For ISFSI and subgrade design parameter Sets A and B. Users may also develop a site-specific lift height limit.

Table 2.2.9

EXAMPLES OF ACCEPTABLE ISFSI PAD DESIGN PARAMETERS

PARAMETER	PARAMETER SET "A" †	PARAMETER SET "B"
Concrete thickness, $t_p$ , (inches)	$\leq 36$	$\leq 28$
Concrete Compressive Strength (at 28 days), $f_c'$ , (psi)	$\leq 4,200$	$\leq 6,000$ psi
Reinforcement Top and Bottom (both directions)	Reinforcing bar shall be 60 ksi Yield Strength ASTM Material	Reinforcing bar shall be 60 ksi Yield Strength ASTM Material
Subgrade Effective Modulus of Elasticity <sup>††</sup> (measured prior to ISFSI pad installation), E, (psi)	$\leq 28,000$	$\leq 16,000$

NOTE: A static coefficient of friction of  $\geq 0.53$  between the ISFSI pad and the bottom of the overpack shall be verified by test. The test procedure shall follow the guidelines included in the Sliding Analysis in Subsection 3.4.7.1.

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

†† An acceptable method of defining the soil effective modulus of elasticity applicable to the drop and tipover analysis is provided in Table 13 of NUREG/CR-6608 with soil classification in accordance with ASTM-D2487 Standard Classification of Soils for Engineering Purposes (Unified Soil Classification System USCS) and density determination in accordance with ASTM-D1586 Standard Test Method for Penetration Test and Split/Barrel Sampling of Soils.

Table 2.2.10  
MPC CONFINEMENT BOUNDARY STRESS INTENSITY LIMITS  
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220)<sup>†</sup>

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

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

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

<sup>†††</sup> No Specific stress limit applicable.

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

Table 2.2.11

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

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

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

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

<sup>†††</sup> No specific stress intensity limit applicable.

Table 2.2.12  
 STRESS LIMITS FOR DIFFERENT  
 LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE OVERPACK AND HI-TRAC  
 (ELASTIC ANALYSIS PER NF-3260)

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

Definitions:

S = Allowable Stress Value for Table 1A, ASME Section II, Part D.

$S_m$  = Allowable Stress Intensity Value from Table 2A, ASME Section II, Part D

$S_u$  = Ultimate Strength

<sup>†</sup> Governed by Appendix F, Paragraph F-1332 of the ASME Code, Section III.

Table 2.2.13

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

NORMAL CONDITION	
LOADING	NOTATION
Dead Weight	D
Handling Loads	H
Design Pressure (Internal) <sup>†</sup>	$P_i$
Design Pressure (External) <sup>†</sup>	$P_o$
Snow	S
Operating Temperature	T
OFF-NORMAL CONDITION	
Loading	Notation
Off-Normal Pressure (Internal) <sup>†</sup>	$P_i'$
Off-Normal Pressure (External) <sup>†</sup>	$P_o'$
Off-Normal Temperature	$T'$
Off-Normal HI-TRAC Handling	$H'$

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



Table 2.2.13 (continued)

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

ACCIDENT CONDITIONS	
LOADING	NOTATION
Handling Accident	H'
Earthquake	E
Fire	T*
Tornado Missile	M
Tornado Wind	W'
Flood	F
Explosion	E*
Accident Pressure (Internal)	P <sub>i</sub> *
Accident Pressure (External)	P <sub>o</sub> *

Table 2.2.14  
 APPLICABLE LOAD CASES AND COMBINATIONS FOR EACH CONDITION AND COMPONENT<sup>†, ††</sup>

CONDITION	LOADING CASE	MPC	OVERPACK	HI-TRAC
Design (ASME Code Pressure Compliance)	1	$P_i, P_o$	N/A	N/A
Normal (Level A)	1	$D, T, H, P_i$	$D, T, H$	$D, T^{\dagger\dagger\dagger}, H, P_{i(\text{water jacket})}$
	2	$D, T, H, P_o$	N/A	N/A
Off-Normal (Level B)	1	$D, T', H, P_i'$	$D, T', H$	$N/A^{\dagger\dagger\dagger}$ (H' pocket trunnion)
	2	$D, T', H, P_o$	N/A	N/A
Accident (Level D)	1	$D, T, P_i, H'$	$D, T, H'$	$D, T, H'$
	2	$D, T^*, P_i^*$	N/A	N/A
	3	$D, T, P_o^{*\dagger\dagger\dagger}$	$D, T, P_o^{*\dagger\dagger\dagger}$	$D, T, P_o^{*\dagger\dagger\dagger}$
	4	N/A	$D, T, (E, M, F, W')^{\dagger\dagger\dagger\dagger}$	$D, T, (M, W')^{\dagger\dagger\dagger\dagger}$

<sup>†</sup> The loading notations are given in Table 2.2.13. Each symbol represents a loading type and may have different values for different components. The different loads are assumed to be additive and applied simultaneously.

<sup>††</sup> N/A stands for “Not Applicable.”

<sup>†††</sup> T (normal condition) for the HI-TRAC is 100°F and  $P_{i(\text{water jacket})}$  is 60 psig and, therefore, there is no off-normal temperature or load combination because Load Case 1, Normal (Level A), is identical to Load Case 1, Off-Normal (Level B). Only the off-normal handling load on the pocket trunnion is analyzed separately.

<sup>††††</sup>  $P_o^*$  bounds the external pressure due to explosion.

<sup>†††††</sup> (E, M, F, W') means loads are considered separately in combination with D, T. E and F not applicable to HI-TRAC.

Table 2.2.15

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
<p><i>MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure.</i></p>	<p><i>Subsection NCA</i></p>	<p><i>General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.</i></p>	<p><i>Because the MPC, overpack, and transfer cask are not ASME Code stamped vessels, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the HI-STORM 100 System as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program.</i></p> <p><i>Because the cask components are not certified to the Code, the terms “Certificate Holder” and “Inspector” are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term “Inspector” means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.</i></p>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
<i>MPC basket supports and lift lugs</i>	<i>NB-1130</i>	<p><i>NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material.</i></p> <p><i>NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.</i></p>	<p><i>The MPC basket supports (nonpressure-retaining structural attachment) and lift lugs (nonstructural attachments (relative to the function of lifting a loaded MPC) that are used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.</i></p>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
<i>MPC, MPC basket assembly, HI-STORM overpack, and HI-TRAC transfer cask</i>	<i>NB-3100 NG-3100 NF-3100</i>	<i>Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.</i>	<i>These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.</i>
MPC	NB-3350	<i>NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.</i>	<p><i>Due to MPC basket-to-shell interface requirements, the MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.</i></p> <p><i>From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.</i></p>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
<i>MPC, MPC basket assembly, HI-STORM overpack steel structure, and HI-TRAC transfer cask steel structure</i>	<i>NB-4120 NG-4120 NF-4120</i>	<i>NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.</i>	<i>In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of lead are not, unless explicitly stated by the Code, defined as heat treatment operations.</i>  <i>For the steel parts in the HI-STORM 100 System components, the duration for which a part exceeds the off-normal temperature limit defined in Chapter 2 of the FSAR shall be limited to 24 hours in a particular manufacturing process (such as the HI-TRAC lead pouring process).</i>
<i>MPC, HI-STORM overpack steel structure, HI-TRAC transfer cask steel structure</i>	<i>NB-4220 NF-4220</i>	<i>Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.</i>	<i>The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-transfer cask) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.</i>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. <del>The MPC vent and drain cover plate welds are leak tested.</del> The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT examination alone is used, at a minimum, it will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	<p>The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be <del>hydrostatically-pressure</del> tested as defined in Chapter 89. Accessibility for leakage inspections preclude a Code compliant <del>hydrostatic-pressure</del> test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded.</p> <p>The inspection <del>process</del> results, including relevant findings (indications), shall be made a permanent part of the user's</p>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
			records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate <del>weld is confirmed by leakage testing and liquid penetrant examination</del> and the closure ring welds <del>are</del> confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	<i>The HI-STORM 100 System is</i> to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NG-2000 requirements.



Table 2.2.15 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

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

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM 100 System <del>will</del> <i>is to</i> be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package <del>will</del> <i>is to</i> be in conformance with Holtec's QA program.
Overpack Steel Structure	NF-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NF-2000 requirements.
HI-TRAC Steel Structure	NF-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NF-2000 requirements.
Overpack Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member thickness of 1" or greater is loaded to transmit loads in the through thickness direction.	The <del>large</del> margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large <del>and warrant an exemption</del> . The overpack baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have <i>large safety factors during lifting</i> . <del>a minimum safety factor of greater than 12 during lifting</del> . <i>Likewise, † the top lid plate to lid shell weld has a large structural margin under the inertia loads imposed during a non-mechanistic tipover event. safety factor greater than 6 under 45g's.</i>

Table 2.2.15 (continued)

## LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
Overpack Steel Structure	NF-3256 NF-3266	Provides requirements for welded joints.	<p>Welds for which no structural credit is taken are identified as “Non-NF” welds on the design drawings by an “N”.</p> <p>These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX.</p> <p><i>Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.</i></p>
<i>HI-STORM Overpack and HI-TRAC Transfer Cask</i>	<i>NF-3320 NF-4720</i>	<i>NF-3324.6 and NF-4720 provide requirements for bolting</i>	<p><i>These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The overpack and transfer cask bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).</i></p> <p><i>Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.</i></p>

Table 2.2.16

COMPARISON BETWEEN HI-STORM MPC LOADINGS WITH HI-STAR MPC LOADINGS<sup>†</sup>

Loading Condition	Difference Between MPC Loadings Under HI-STAR and HI-STORM Conditions
Dead Load	Unchanged
Design Internal Pressure (normal, off-normal, & accident)	Unchanged
Design External Pressure (normal, off-normal, & accident)	HI-STORM normal and off-normal external pressure is ambient which is less than the HI-STAR 40 psig. The accident external pressure is unchanged.
Thermal Gradient (normal, off-normal, & accident)	Determined by analysis in Chapters 3 and 4
Handling Load (normal)	Unchanged
Earthquake (accident)	Inertial loading increased less than 0.1g's (for free-standing overpack designs).
Handling Load (accident)	HI-STORM vertical and horizontal deceleration loadings are less than those in HI-STAR, but the HI-STORM cavity inner diameter is different and therefore the horizontal loading on the MPC is analyzed in Chapter 3.

<sup>†</sup> HI-STAR MPC loadings are those specified in HI-STAR SARs under Docket Numbers 71-9261 and 72-1008.

## 2.3 SAFETY PROTECTION SYSTEMS

### 2.3.1 General

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

1. The MPC confinement barrier is an enclosure vessel designed in accordance with the ASME Code, Subsection NB with confinement welds inspected by radiography (RT) or ultrasonic testing (UT). Where RT or UT is not possible, a redundant closure system is provided with field welds which are hydrostatically pressure tested, ~~helium leakage tested~~ and/or inspected by the liquid penetrant method (see Section 9.1).
2. The MPC confinement barrier is surrounded by the HI-STORM overpack which provides for the physical protection of the MPC.
3. The HI-STORM 100 System is designed to meet the requirements of storage while maintaining the safety of the SNF.
4. The SNF once initially loaded in the MPC does not require opening of the canister for repackaging to transport the SNF.
5. The decay heat emitted by the SNF is rejected from the HI-STORM 100 System through passive means. No active cooling systems are employed.

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

The structures, systems, and components of the HI-STORM 100 System designated as important to safety are identified in Table 2.2.6. Similar categorization of structures, systems, and components, which are part of the ISFSI, but not part of the HI-STORM 100 System, will be the responsibility of the 10CFR72 licensee. For HI-STORM 100A, the ISFSI pad is designated ITS, Category C as discussed in Subsection 2.0.4.1.

## 2.3.2 Protection by Multiple Confinement Barriers and Systems

### 2.3.2.1 Confinement Barriers and Systems

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

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

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

The HI-STORM 100 System is designed with several confinement barriers for the radioactive fuel contents. Intact fuel assemblies have cladding which provides the first boundary preventing release of the fission products. Fuel assemblies classified as damaged fuel or fuel debris are placed in a damaged fuel container which restricts the release of fuel debris. The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, shell, lid, closure ring, and port cover plates.

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

Confinement is discussed further in Chapter 7. MPC field weld examinations, ~~hydrostatic and pressure testing, and helium leak testing~~ are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 9, to verify the confinement boundary.

### 2.3.2.2 Cask Cooling

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

- The heat rejection capacity of the HI-STORM 100 System is deliberately understated by conservatively determining the design basis fuel *that maximizes thermal resistance (see Section 2.1.6)*. ~~The decay heat value in Table 2.1.6 is developed by computing the decay heat from the design basis fuel assembly which produces the highest heat generation rate for a given burnup.~~ Additional margin is built into the calculated cask cooling rate by using ~~at~~ the design basis fuel assembly ~~that~~ *which* offers maximum resistance to *MPC internal helium circulation* ~~the transmission of heat (minimum thermal conductivity)~~.
- The MPC fuel basket is formed by a honeycomb structure of stainless steel plates with full-length edge-welded intersections, which allows the unimpaired conduction of heat.
- The MPC confinement boundary ensures that the helium atmosphere inside the MPC is maintained during normal, off-normal, and accident conditions of storage and transfer. The MPC confinement boundary maintains the helium confinement atmosphere below the design temperatures and pressures stated in Table 2.2.3 and Table 2.2.1, respectively.
- The MPC thermal design maintains the fuel rod cladding temperatures below the values stated in Chapter 4 such that fuel cladding is not degraded during the long term storage period.
- The HI-STORM is optimally designed with cooling vents and an MPC to overpack annulus which maximize air flow, while providing superior radiation shielding. The vents and annulus allow cooling air to circulate past the MPC removing the decay heat.

### 2.3.3 Protection by Equipment and Instrumentation Selection

#### 2.3.3.1 Equipment

Design criteria for the HI-STORM 100 System are described in Section 2.2. The HI-STORM 100 System may include use of ancillary or support equipment for ISFSI implementation.

Ancillary equipment and structures utilized outside of the reactor facility's 10CFR Part 50 structures may be broken down into two broad categories, namely Important to Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", provides guidance for the determination of a component's safety classification. Certain ancillary equipment (such as trailers, rail cars, skids, portable cranes, transporters, or air pads) are not required to be designated as ITS for most ISFSI implementations, if the HI-STORM 100 is designed to withstand the failure of these components.

The listing and ITS designation of ancillary equipment in Table 8.1.6 follows NUREG/CR-6407. ITS ancillary equipment utilized in activities that occur outside the 10CFR Part 50 structure shall be engineered to meet all functional, strength, service life, and operational safety requirements to ensure that the design and operation of the ancillary equipment is consistent with the intent of this Safety Analysis Report. The design for these components shall consider the following information, as applicable:

1. Functions and boundaries of the ancillary equipment
2. The environmental conditions of the ISFSI site, including tornado-borne missile, tornado wind, seismic, fire, lightning, explosion, ambient humidity limits, flood, tsunami and any other environmental hazards unique to the site.
3. Material requirements including impact testing requirements
4. Applicable codes and standards
5. Acceptance testing requirements
6. Quality assurance requirements
7. Foundation type and permissible loading
8. Applicable loads and load combinations
9. Pre-service examination requirements
10. In-use inspection and maintenance requirements
11. Number and magnitude of repetitive loading significant to fatigue
12. Insulation and enclosure requirements (on electrical motors and machinery)
13. Applicable Reg. Guides and NUREGs.
14. Welding requirements
15. Painting, marking, and identification requirements
16. Design Report documentation requirements
17. Operational and Maintenance (O&M) Manual information requirements

All design documentation shall be subject to a review, evaluation, and safety assessment process in accordance with the provisions of the QA program described in Chapter 13.

Users may effectuate the inter-cask transfer of the MPC between the HI-TRAC transfer cask and either the HI-STORM 100 or the HI-STAR 100 overpack in a location of their choice, depending upon site-specific needs and capabilities. For those users choosing to perform the MPC inter-cask transfer *using devices not integral to structures outside of a facility* governed by the regulations of 10 CFR Part 50 (e.g., fuel handling or reactor building), a Cask Transfer



Facility (CTF) is required. The CTF is a stand-alone facility located on-site, near the ISFSI that incorporates or is compatible with lifting devices designed to lift a loaded or unloaded HI-TRAC transfer cask, place it atop the overpack, and transfer the loaded MPC to or from the overpack. The detailed design criteria which must be followed for the design and operation of the CTF are set down in Paragraphs A through R below.

The inter-cask transfer operations consist of the following potential scenarios of MPC transfer:

- Transfer between a HI-TRAC transfer cask and a HI-STORM overpack
- Transfer between a HI-TRAC transfer cask and a HI-STAR 100 overpack

In both scenarios, the standard design HI-TRAC is mounted on top of the overpack (HI-STAR 100, HI-STORM 100, HI-STORM 100S) and the MPC transfer is carried out by opening the transfer lid doors located at the bottom of the HI-TRAC transfer cask and by moving the MPC vertically to the cylindrical cavity of the recipient cask. For the HI-TRAC 125D design, the MPC transfer is carried out in a similar fashion, except that there is no transfer lid involved - the pool lid is removed while the transfer cask is mounted atop the HI-STORM overpack with the HI-STORM mating device located between the two casks (see Figure 1.2.18). However, the devices utilized to lift the HI-TRAC cask to place it on the overpack and to vertically transfer the MPC may be of stationary or mobile type.

The specific requirements for the CTF employing stationary and mobile lifting devices are somewhat different. The requirements provided in the following specification for the CTF apply to both types of lifting devices, unless explicitly differentiated in the text.

A. General Specifications:

- i. The cask handling functions which may be required of the Cask Transfer Facility include:
  - a. Upending and downending of a HI-STAR 100 overpack on a flatbed rail car or other transporter (see Figure 2.3.1 for an example).
  - b. Upending and downending of a HI-TRAC transfer cask on a heavy-haul transfer trailer or other transporter (see Figure 2.3.2 for an example)
  - c. Raising and placement of a HI-TRAC transfer cask on top of a HI-STORM 100 overpack for MPC transfer operations (see Figure 2.3.3 for an example of the cask arrangement with the standard design HI-TRAC transfer cask. The HI-TRAC 125D design would include the mating device and no transfer lid).

- d. Raising and placement of a HI-TRAC transfer cask on top of a HI-STAR 100 overpack for MPC transfer operations (see Figure 2.3.4 for an example of the cask arrangement with the standard design HI-TRAC transfer cask. The HI-TRAC 125D design would include the mating device and no transfer lid).
  - e. MPC transfer between the HI-TRAC transfer cask and the HI-STORM overpack.
  - f. MPC transfer between the HI-TRAC transfer cask and the HI-STAR 100 overpack.
- ii. Other Functional Requirements:

The CTF should possess facilities and capabilities to support cask operations such as :

- a. Devices and areas to support installation and removal of the HI-STORM overpack lid.
- b. Devices and areas to support installation and removal of the HI-STORM 100 overpack vent shield block inserts.
- c. Devices and areas to support installation and removal of the HI-STAR 100 closure plate.
- d. Devices and areas to support installation and removal of the HI-STAR 100 transfer collar.
- e. Features to support positioning and alignment of the HI-STORM overpack and the HI-TRAC transfer cask.
- f. Features to support positioning and alignment of the HI-STAR 100 overpack and the HI-TRAC transfer cask.
- g. Areas to support jacking of a loaded HI-STORM overpack for insertion of a translocation device underneath.
- h. Devices and areas to support placement of an empty MPC in the HI-TRAC transfer cask or HI-STAR 100 overpack
- i. Devices and areas to support receipt inspection of the MPC, HI-TRAC transfer cask, HI-STORM overpack, and HI-STAR overpack.

- j. Devices and areas to support installation and removal of the HI-STORM mating device (HI-TRAC 125D only).

iii. Definitions:

The components of the CTF covered by this specification consist of all structural members, lifting devices, and foundations which bear all or a significant portion of the dead load of the transfer cask or the multi-purpose canister during MPC transfer operations. The definitions of key terms not defined elsewhere in this FSAR and used in this specification are provided below. The following terms are used to define key components of the CTF.

- Connector Brackets: The mechanical part used in the load path which connects to the cask trunnions. A fabricated weldment, slings, and turnbuckles are typical examples of connector brackets.
- CTF structure: The CTF structure is the stationary, anchored portion of the CTF which provides the required structural function to support MPC transfer operations, including lateral stabilization of the HI-TRAC transfer cask and, if required, the overpack, to protect against seismic events. The MPC lifter, if used in the CTF design, is integrated into the CTF structure (see Lifter Mount).
- HI-TRAC lifter(s): The HI-TRAC lifter is the mechanical lifting device, typically consisting of jacks or hoists, that is utilized to lift a loaded or unloaded HI-TRAC to the required elevation in the CTF so that it can be mounted on the overpack.<sup>†</sup>
- Lifter Mount: A beam-like structure (part of the CTF structure) that supports the HI-TRAC and MPC lifter(s).
- Lift Platform: The lift platform is the intermediate structure that transfers the vertical load of the HI-TRAC transfer cask to the HI-TRAC lifters.

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<sup>†</sup> The term overpack is used in this specification as a generic term for the HI-STAR 100 and the various HI-STORM overpacks.

- Mobile crane: A mobile crane is a device defined in ASME B30.5-1994, Mobile and Locomotive Cranes. A mobile crane may be used in lieu of the HI-TRAC lifter and/or an MPC lifter provided all requirements set forth in this subsection are satisfied.
- MPC lifter: The MPC lifter is a mechanical lifting device, typically consisting of jacks or hoists, that is utilized to vertically transfer the MPC between the HI-TRAC transfer cask and the overpack.
- Pier: The portion of the reinforced concrete foundation which projects above the concrete floor of the CTF.
- Single-Failure-Proof (SFP): A single-failure-proof handling device is one wherein all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria given in of NUREG-0612.
- Translocation Device: A low vertical profile device used to laterally position an overpack such that the bottom surface of the overpack is fully supported by the top surface of the device. Typical translocation devices are air pads and Hillman rollers.

iv. Important to Safety Designation:

All components and structures which comprise the CTF shall be given an ITS category designation in accordance with a written procedure which is consistent with NUREG/CR-6407 and ~~Chapter 13 of this FSAR~~ *the Holtec quality assurance program.*

B. Environmental and Design Conditions

- i. Lowest Service Temperature (LST): The LST for the CTF is 0°F (consistent with the specification for the HI-TRAC transfer cask in Subsection 3.1.2.3).
- ii. Snow and Ice Load, S: The CTF structure shall be designed to withstand the dead weight of snow and ice for unheated structures as set forth in ASCE 7-88 [2.2.2] for the specific ISFSI site.

- iii. Tornado Missile, M, and Tornado Wind, W': The tornado wind and tornado-generated missile data applicable to the HI-STORM 100 System (Tables 2.2.4 and 2.2.5) will be used in the design of the CTF structure unless existing site design basis data or a probabilistic risk assessment (PRA) for the CTF site with due consideration of short operation durations indicates that a less severe tornado missile impact or wind loading on the CTF structure can be postulated. The PRA analysis can be performed in the manner of the EPRI Report NP-2005, "Tornado Missile Simulation and Design Methodology Computer Code Manual". USNRC Reg. Guide 1.117 and Section 2.2.3 of NUREG-800 may be used for guidance in establishing the appropriate tornado missile and wind loading for the CTF structure.

The following additional clarifications apply to the large tornado missile (4,000 lb. automobile) in Tables 2.2.4 and 2.2.5 in the CTF structure analysis:

- The missile has a planform area of 20 sq. ft. and impact force characteristics consistent with the *HI-TRAC missile impact analysis*. ~~set forth in Appendix 3.AN (Section 3.AN.3).~~
- The large missile can strike the CTF structure in any orientation up to an elevation of 15 feet.

If the site tornado missile data developed by the ISFSI owner suggests that tornado missiles of greater kinetic energies than that postulated in this FSAR (Table 2.2.4 and 2.2.5) should be postulated for CTF during its use, then the integrity analysis of the CTF structure shall be carried out under the site-specific tornado missiles. This situation would also require the HI-TRAC transfer cask and the overpack to be re-evaluated under the provisions of 10CFR72.212 and 72.48.

The wind speed specified in this FSAR (Tables 2.2.4 and 2.2.5), likewise, shall be evaluated for their applicability to the site. Lower or higher site-specific wind velocity, compared to the design basis values cited in this FSAR shall be used if justified by appropriate analysis, which may include PRA.

Intermediate penetrant missile and small missiles postulated in this FSAR are not considered to be a credible threat to the functional integrity of the CTF structure and, therefore, need not be considered.

- iv. Flood: The CTF will be assumed to be flooded to the highest elevation for the CTF facility determined from the local meteorological data. The flood velocity shall be taken as the largest value defined for the ISFSI site.
- v. Lightning: Meteorological data for the region surrounding the ISFSI site shall be used to specify the applicable lightning input to the CTF structure for personnel safety evaluation purposes.
- vi. Water Waves (Tsunami, Y): Certain coastal CTF sites may be subject to sudden, short duration waves of water, denoted in the literature by various terms, such as tsunami. If the applicable meteorological data for the CTF site indicates the potential of such water-borne loadings on the CTF structure, then such a loading, with due consideration of the short duration of CTF operations, shall be defined for the CTF structure.
- vii. Design Basis Earthquake (DBE), E: The DBE event applicable to the CTF facility pursuant to 10CFR100, Appendix A, shall be specified. The DBE should be specified as a set of response spectra or acceleration time-histories for use in the CTF structural and impact consequence analyses.
- viii. Design Temperature: All material properties used in the stress analysis of the CTF structure shall utilize a reference design temperature of 150°F.

C. Heavy Load Handling:

- i. Apparent dead load, D\*: The dead load of all components being lifted shall be increased in the manner set forth in Subsection 3.4.3 to define the Apparent Dead Load, D\*.
- ii. NUREG-0612 Conformance:

The Connector Bracket, HI-TRAC lifter, and MPC lifter shall comply with the guidance provided in NUREG-0612 (1980) for single failure proof devices. Where the geometry of the lifting device is different from the configurations contemplated by NUREG-0612, the following exceptions apply:

- a. Mobile cranes at the CTF shall conform to the guidelines of Section 5.1.1 of NUREG-0612 with the exception that mobile

cranes shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes", in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes". The mobile crane used shall have a minimum safety factor of two over the allowable load table for the crane in accordance with Section 5.1.6(1)(a) of NUREG-0612, and shall be capable of stopping and holding the load during a DBE event.

- b. Section 5.1.6(2) of NUREG-0612 specifies that new cranes should be designed to meet the requirements of NUREG-0554. For mobile cranes, the guidance of Section 5.1.6(2) of NUREG-0612 does not apply.
- iii. Defense-in-Depth Measures:
  - a. The lift platform and the lifter mount shall be designed to ensure that the stresses produced under the apparent dead load,  $D^*$ , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.
  - b. The CTF structure shall be designed to ensure that the stresses produced in it under the apparent dead load,  $D^*$ , are less than the Level A (normal condition) stress limits for ASME Section III, Subsection NF, Class 3, linear structures.
  - c. Maximum deflection of the lift platform and the lifter mount under the apparent dead load shall comply with the limits set forth in CMAA-70.
  - d. When the HI-TRAC transfer cask is stacked on the overpack, HI-TRAC shall be either held by the lifting device or laterally restrained by the CTF structure. Furthermore, when the HI-TRAC transfer cask is placed atop the overpack, the overpack shall be laterally restrained from uncontrolled movement, if required by the analysis specified in Subsection 2.3.3.1.N.
  - e. The design of the lifting system shall ensure that the lift platform (or lift frame) is held horizontal at all times and that the symmetrically situated axial members are symmetrically loaded.

- f. In order to minimize occupational radiation exposure to ISFSI personnel, design of the MPC lifting attachment (viz., sling) should not require any human activity inside the HI-TRAC cylindrical space.
  - g. The HI-TRAC lifter and MPC lifter shall possess design features to avoid side-sway of the payload during lifting operations.
  - h. The lifter (HI-TRAC and MPC) design shall ensure that any electrical malfunction in the motor or the power supply will not lead to an uncontrolled lowering of the load.
  - i. The kinematic stability of HI-TRAC or HI-STORM standing upright in an unrestrained configuration (if such a condition exists during the use of the CTF) shall be analytically evaluated and ensured under all postulated extreme environmental phenomena loadings for the CTF facility.
- iv. Shielding Surety:

The design of the HI-TRAC and MPC lifters shall preclude the potential for the MPC to be removed, completely or partially, from the cylindrical space formed by the HI-TRAC and the underlying overpack.

- v. Specific Requirements for Mobile Cranes:

A mobile crane, if used in the CTF in the role of the HI-TRAC lifter or MPC lifter is governed in part by ANSI/ASME N45.2.15 with technical requirements specified in ANSI B30.5 (1994).

When lifting the MPC from an overpack to the HI-TRAC transfer cask, limit switches or load limiters shall be set to ensure that the mobile crane is prevented from lifting loads in excess of 110% of the loaded MPC weight.

An analysis of the consequences of a potential MPC vertical drop which conforms to the guidelines of Appendix A to NUREG-0612 shall be performed. The analysis shall demonstrate that a postulated drop would not result in the MPC experiencing a deceleration in excess of its design basis deceleration specified in this FSAR.



- vi. Lift Height Limitation: The HI-TRAC lift heights shall be governed by the Technical Specifications.
- vii. Control of Side Sway: Procedures shall provide provisions to ensure that the load is lifted essentially vertically with positive control of the load. Key cask lifting and transfer procedures, as determined by the user, should be reviewed by the Certificate Holder before their use.

D. Loads and Load Combinations for the CTF Structure

The applicable loadings for the CTF have been summarized in paragraph B in the preceding. A stress analysis of the CTF structure shall be performed to demonstrate compliance with the Subsection NF stress limits for Class 3 linear structures for the service condition germane to each load combination. Table 2.3.2 provides the load combinations (the symbols in Table 2.3.2 are defined in the preceding text and in Table 2.2.13).

E. Materials and Failure Modes

- i. Acceptable Materials and Material Properties: All materials used in the design of the CTF shall be ASTM approved or equal, consistent with the ITS category of the part. Reinforced concrete, if used, shall comply with the provisions of ACI 318 (89). The material property and allowable stress values for all steel structurals shall be taken from the ASME and B&PV Code, Section II, wherever such data is available; otherwise, the data provided in the ASTM standards shall be used.
- ii. Brittle Fracture: All structural components in the CTF structure and the lift platform designated as primary load bearing shall have an NDTT equal to 0°F or lower (consistent with the ductile fracture requirements for ASME Section III, Subsection NF, Class 3 structures).
- iii. Fatigue: Fatigue failure modes of primary structural members in the CTF structure whose failure may result in uncontrolled lowering of the HI-TRAC transfer cask or the MPC (critical members) shall be evaluated. A minimum factor of safety of 2 on the number of permissible loading cycles on the critical members shall apply.

- iv. Buckling: For all critical members in the CTF structure (defined above), potential failure modes through buckling under axial compression shall be considered. The margin of safety against buckling shall comply with the provisions of ASME Section III, Subsection NF, for Class 3 linear structures.

F. CTF Pad

A reinforced concrete pad in conformance with the specification for the ISFSI pad set forth in this FSAR (see Table 2.2.9) may be used in the region of the CTF where the overpack and HI-TRAC are stacked for MPC transfer. Alternatively, the pad may be designed using the guidelines of ACI-318(89).

G. Miscellaneous Components

Hoist rings, turnbuckles, slings, and other appurtenances which are in the load path during heavy load handling at the CTF shall be single-failure-proof.

H. Structural Welds

All primary structural welds in the CTF structure shall comply with the specifications of ASME Section III for Class 3 NF linear structures.

I. Foundation

The design of the CTF structure foundation and piers, including load combinations, shall be in accordance with ACI-318(89).

J. Rail Access

The rail lines that enter the Cask Transfer Facility shall be set at grade level with no exposed rail ties or hardware other than the rail itself.

K. Vertical Cask Crawler/Translocation Device Access (If Required)

- i. The cask handling bay in the CTF shall allow access of a vertical cask crawler or translocation device carrying a transfer cask or overpack. The building floor shall be equipped with a smooth transition to the cask travel route such that the vertical cask crawler tracks do not have to negotiate sharp lips or slope transitions and the translocation devices have a smooth transition. Grading of exterior aprons shall be no more than necessary to allow water drainage.

- ii. If roll-up doors are used, the roll up doors shall have no raised threshold that could damage the vertical cask crawler tracks (if a crawler is used).
- iii. Exterior aprons shall be of a material that will not be damaged by the vertical cask crawler tracks, if a crawler is used.

L. Facility Floor

- i. The facility floor shall be sufficiently flat to allow optimum handling of casks with a translocation device.
- ii. Any floor penetrations, in areas where translocation device operations may occur, shall be equipped with flush inserts.
- iii. The rails, in areas where translocation device operations may occur shall be below the finish level of the floor. Flush inserts, if necessary, shall be sized for installation by hand.

M. Cask Connector Brackets

- i. Primary lifting attachments between the cask and the lifting platform are the cask connector brackets. The cask connector brackets may be lengthened or shortened to allow for differences in the vehicle deck height of the cask delivery vehicle and the various lifting operations. The connector brackets shall be designed to perform cask lifting, upending and downending functions. The brackets shall be designed in accordance with ANSI N14.6 [Reference 2.2.3] and load tested at 300% of the load applied to them during normal handling.
- ii. The connector brackets shall be equipped with a positive engagement to ensure that the cask lifting attachments do not become inadvertently disconnected during a seismic event and during normal cask handling operations.
- iii. The design of the connector brackets shall ensure that the HI-TRAC transfer cask is fully secured against slippage during MPC transfer operations.

N. Cask Restraint System

A time-history analysis of the stacked overpack/HI-TRAC transfer cask assemblage under the postulated ISFSI Level D events in Table 2.3.2 shall be performed to demonstrate that a minimum margin of safety of 1.1 against overturning or kinematic instability exists and that the CTF structure complies with the applicable stress limits (Table 2.3.2) and that the maximum permissible deceleration loading specified in the FSAR is not exceeded. If required to meet the minimum margin of safety of 1.1, a cask restraining system shall be incorporated into the design of the Cask Transfer Facility to provide lateral restraint to the overpack (HI-STORM or HI-STAR 100).

O. Design Life

The Cask Transfer Facility shall be constructed to have a minimum design life of 40 years.

P. Testing Requirements

In addition to testing recommended in NUREG-0612 (1980), a structural adequacy test of the CTF structure at 125% of its operating load prior to its first use in a cask loading campaign shall be performed. This test should be performed in accordance with the guidance provided in the CMAA Specification 70 [2.2.16].

Q. Quality Assurance Requirements

All components of the CTF shall be manufactured in full compliance with the quality assurance requirements applicable to the ITS category of the component as set forth in *the Holtec QA program. Chapter 13 of this FSAR.*

R. Documentation Requirements

- i. O&M Manual: An Operations and Maintenance Manual shall be prepared which contains, at minimum, the following items of information:
  - Maintenance Drawings
  - Operating Procedures
- ii. Design Report: A QA-validated design report documenting full compliance with the provisions of this specification shall be prepared and archived for future reference in accordance with the provisions of

### 2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STORM 100 System, instrumentation which is important to safety is not necessary. No instrumentation is required or provided for HI-STORM 100 storage operations, other than normal security service instruments and TLDs.

However, in lieu of performing the periodic inspection of the HI-STORM overpack vent screens, temperature elements may be installed in two of the overpack exit vents to continuously monitor the air temperature. If the temperature elements and associated temperature monitoring instrumentation are used, they shall be designated important to safety as specified in Table 2.2.6.

The temperature elements and associated temperature monitoring instrumentation provided to monitor the air outlet temperature shall be suitable for a temperature range of -40°F to 500°F. At a minimum, the temperature elements and associated temperature monitoring instrumentation shall be calibrated for the temperatures of 32°F (ice point), 212°F (boiling point), and 449°F (melting point of tin) with an accuracy of +/- 4°F.

### 2.3.4 Nuclear Criticality Safety

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

#### 2.3.4.1 Control Methods for Prevention of Criticality

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

- a. Incorporation of permanent neutron absorbing material (~~Boral~~<sup>Boral™</sup>) in the MPC fuel basket walls.
- b. Favorable geometry provided by the MPC fuel basket

Additional control methods used to prevent criticality for the MPC-24, MPC-24E, and MPC-24EF (all with higher enriched fuel), and the MPC-32 and MPC-32F are the following:

- a. Loading of PWR fuel assemblies must be performed in water with a minimum boron content as specified in Table 2.1.14 or 2.1.16, as applicable.
- b. Prevention of fresh water entering the MPC internals.

~~Administrative controls specified as Technical Specifications and Approved Contents are provided in Appendices A and B to the CoC, respectively, and shall be used to ensure that fuel placed in the HI-STORM 100 System meets the requirements described in Chapters 2 and 6. All appropriate criticality analyses are presented in Chapter 6.~~

#### 2.3.4.2 Error Contingency Criteria

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

#### 2.3.4.3 Verification Analyses

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

#### 2.3.5 Radiological Protection

##### 2.3.5.1 Access Control

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

##### 2.3.5.2 Shielding

The shielding design is governed by 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive doses in excess of the limits given in Table 2.3.1 for normal, off-normal, and accident conditions.

The objective of shielding is to assure that radiation dose rates at key locations are *as low as practical in order to maintain occupational doses to operating personnel As Low As Reasonably Achievable (ALARA) and to meet the requirements of 10 CFR 72.104 and 10 CFR 106 for dose at the controlled area boundary.* ~~below acceptable levels for those locations.~~ Three locations are of particular interest in the storage mode:

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

Dose rates in the immediate vicinity of the loaded overpack are important in consideration of occupational exposure. *Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on the contents of the BWR and PWR MPCs permitted for storage as described in Section 2.1.9. Actual dose rates in operation will be lower than those reported in Chapter 5 for the following reasons:*

- *The shielding evaluation model has a number of conservatisms, as discussed in Chapter 5.*
- *No single cask will likely contain design basis fuel in each fuel storage location and the full compliment of non-fuel hardware allowed by the CoC.*
- *No single cask will contain fuel and non-fuel hardware at the limiting burnups and cooling times allowed by the CoC.*

*Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM 100 System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask* A design objective for the maximum average radial surface dose rate has been established as ~~60-100~~ 100 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 60 mrem/hr. The average dose rate at the top of the overpack is limited to below 60 mrem/hr. Chapter 5 of this FSAR presents the analyses and evaluations to establish HI-STORM 100 compliance with these design objectives.

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

For the loading and unloading of the HI-STORM overpack with the MPC, three transfer cask designs are provided (i.e., HI-TRAC 125, HI-TRAC 100, and HI-TRAC 125D). The two 125 ton HI-TRAC provide better shielding than the 100 ton HI-TRAC due to the increased shielding

thickness and corresponding greater weight. Provided the licensee is capable of utilizing the 125 ton HI-TRAC, ALARA considerations would normally dictate that the 125 ton HI-TRAC should be used. However, sites may not be capable of utilizing the 125 ton HI-TRAC due to crane capacity limitations, floor loading limitations, or other site-specific considerations. As with other dose reduction-based plant activities, , individual users who cannot accommodate the 125 ton HI-TRAC should perform a cost-benefit analysis of the actions (e.g., plant modifications) that would be necessary to use the 125 ton HI-TRAC. The cost of the action(s) would be weighed against the value of the projected reduction in radiation exposure and a decision made based on each plant's particular ALARA implementation philosophy.

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

The analyses presented in Chapters 5, 10, and 11 demonstrate that the HI-STORM 100 System *is capable of meeting* the above radiation dose limits ~~and design objectives~~.

#### 2.3.5.3 Radiological Alarm System

There are no credible events ~~which-that~~ could result in release of radioactive materials or increases in direct radiation above the requirements of 10CFR72.106. ~~In addition, the non-mechanistic release as the result of a hypothetical accident is described in Chapter 7, and results in a dose to an individual at the controlled area boundary of a very small magnitude. Therefore, radiological alarm systems are not necessary.~~

#### 2.3.6 Fire and Explosion Protection

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

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



Table 2.3.1

RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

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

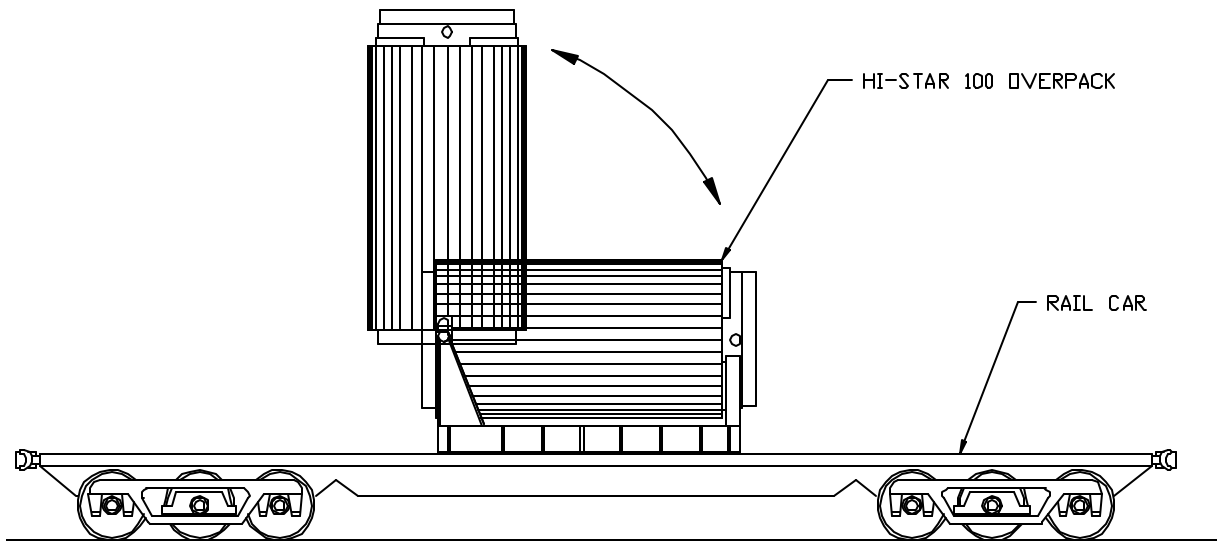
Table 2.3.2

Load Combinations<sup>†</sup> and Service Condition Definitions for the CTF Structure

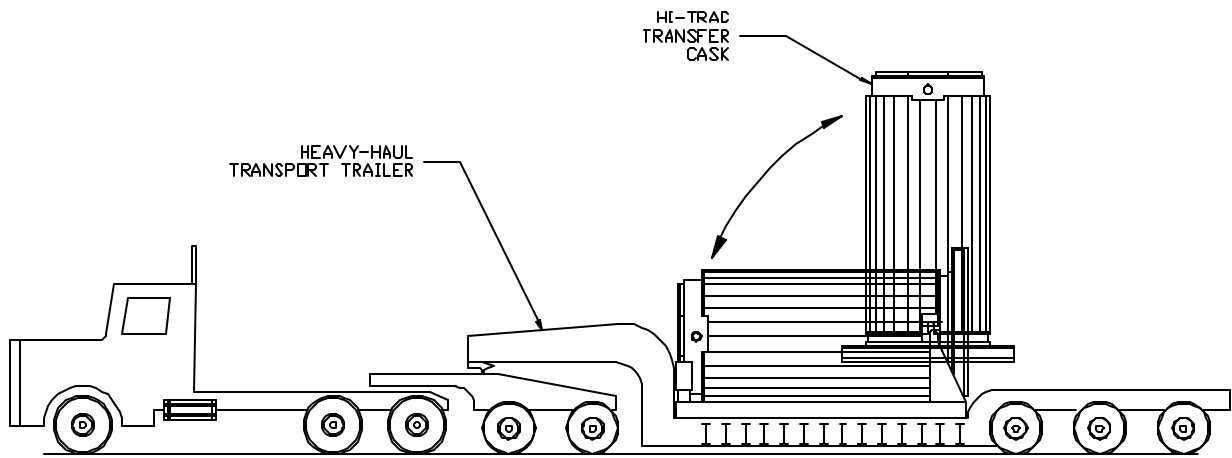
Load Combination	Service Condition for Section III of the ASME Code for Definition of Allowable Stress	Comment
D*	Level A	All primary load bearing members must satisfy Level A stress limits.
D+S	Level A	
D+M <sup>††</sup> +W' D+F D+E or D+Y	Level D	Factor of safety against overturning shall be $\geq 1.1$

<sup>†</sup> The reinforced concrete portion of the CTF structure shall also meet factored combinations of the above loads set forth in ACI-318(89).

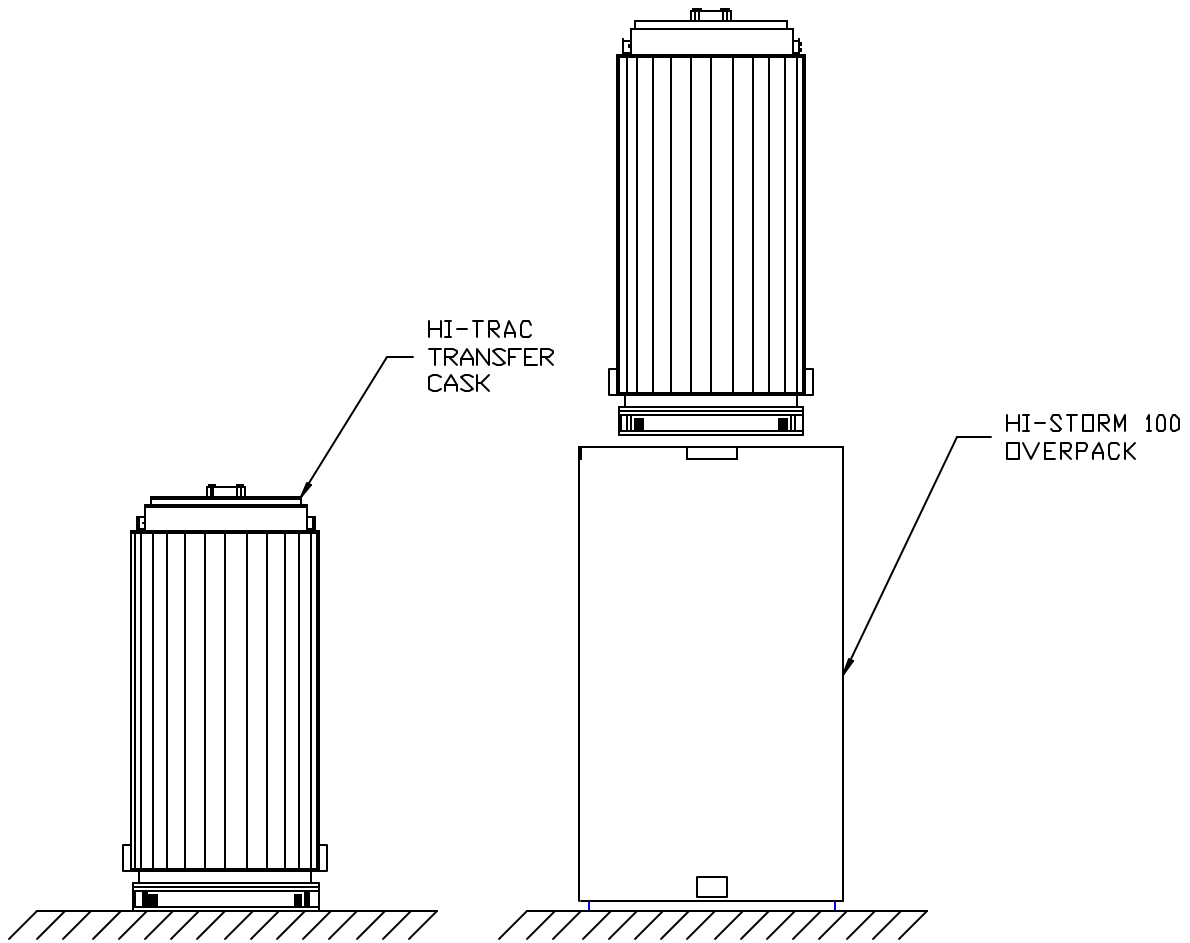
<sup>††</sup> This load may be reduced or eliminated based on a PRA for the CTF site.



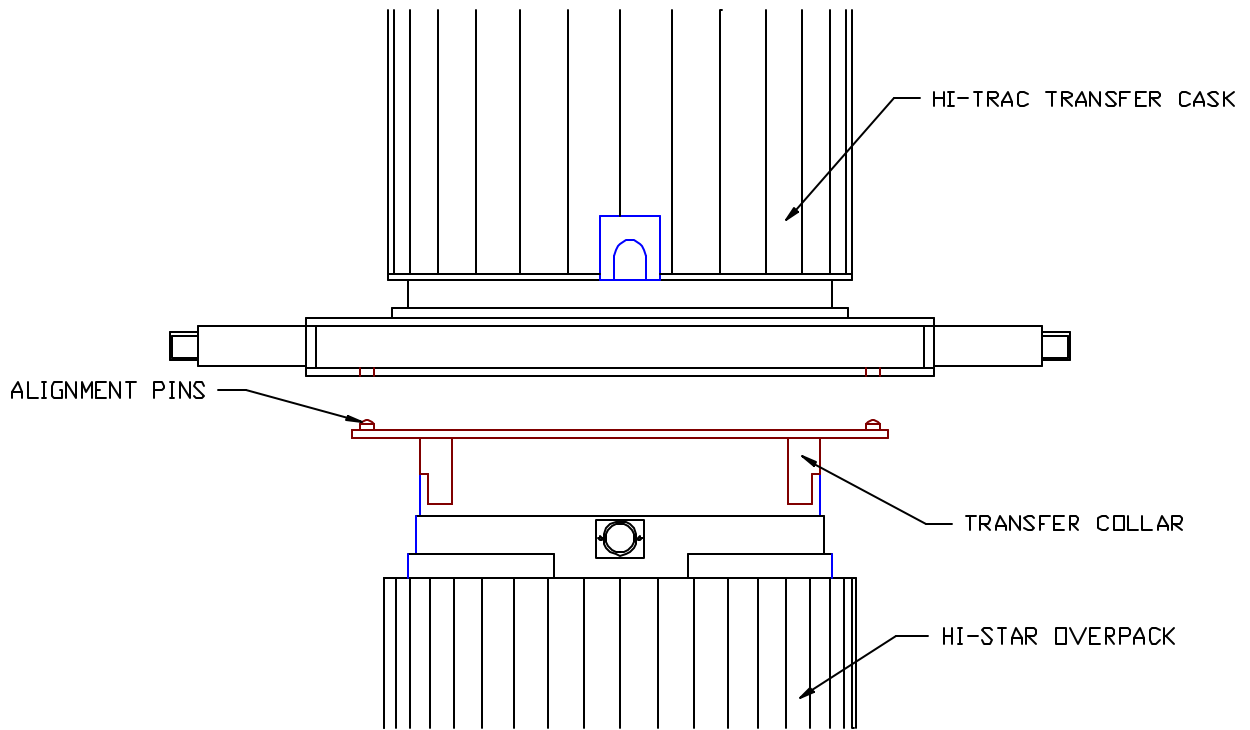
**FIGURE 2.3.1; HI-STAR 100 UPENDING AND DOWNENDING ON A RAIL CAR**



**FIGURE 2.3.2; HI-TRAC UPENDING AND DOWNENDING ON A HEAVY-HAUL TRANSPORT TRAILER**



**FIGURE 2.3.3; HI-TRAC PLACEMENT ON HI-STORM 100 FOR MPC TRANSFER OPERATIONS**



**FIGURE 2.3.4; HI-TRAC PLACEMENT ON HI-STAR 100 FOR MPC TRANSFER OPERATIONS**

## 2.4 DECOMMISSIONING CONSIDERATIONS

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

The MPC is being licensed for transport off-site in the HI-STAR 100 dual-purpose cask system (Reference Docket No. 71-9261). No further handling of the SNF stored in the MPC is required prior to transport to a licensed centralized storage facility or licensed repository.

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

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

In either case, the overpack would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

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

It is also likely that both the overpack and MPC, or extensive portions of both, can be further decontaminated to allow recycle or reuse options. After decontamination, the only radiological hazard the HI-STORM 100 System may pose is slight activation of the HI-STORM 100 materials caused by irradiation over a 40-year storage period.

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

To evaluate the effects on the MPC and HI-STORM overpack caused by irradiation over a 40-year storage period, the following analysis is provided. Table 2.4.1 provides the conservatively determined quantities of the major nuclides after 40 years of irradiation. The calculation of the material activation is based on the following:

- Beyond design basis fuel assemblies (B&W 15x15, ~~3-74.8%~~ enrichment, ~~47,500~~70,000 MWD/MTU, and ~~eight~~five-year cooling time) stored for 40 years. *A constant source term for 40 years was used with no decrease in the neutron source term. This bounds the source term associated with the limiting PWR burnup of 68,200 MWD/MTU.*
- Material quantities based on the ~~Design D~~drawings in Section 1.5.
- A constant flux equal to the initial loading condition is conservatively assumed for the full 40 years.
- Material activation is based on MCNP-4A calculations.

As can be seen from the material activation results presented in Table 2.4.1, the MPC and HI-STORM overpack activation is very low, even including the very conservative assumption of a constant flux for 40 years. The results for the concrete in the HI-STORM overpack can be conservatively applied to the ISFSI pad. This is extremely conservative because the overpack shields most of the flux from the fuel and, therefore, the ISFSI pad will experience a minimal flux.

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



Table 2.4.1  
MPC ACTIVATION

<b>Nuclide</b>	<b>Activity After 40-Year Storage (Ci/m<sup>3</sup>)</b>
<sup>54</sup> Mn	2.20e-3
<sup>55</sup> Fe	3.53e-3
<sup>59</sup> Ni	2.91e-6
<sup>60</sup> Co	3.11e-4
<sup>63</sup> Ni	9.87e-5
<b>Total</b>	<b>6.15e-3</b>

HI-STORM OVERPACK ACTIVATION

<b>Nuclide</b>	<b>Activity After 40-Year Storage (Ci/m<sup>3</sup>)</b>
Overpack Steel	
<sup>54</sup> Mn	3.62e-4
<sup>55</sup> Fe	7.18e-3
<b>Total</b>	<b>7.18e-3</b>
Overpack Concrete	
<sup>39</sup> Ar	3.02e-6
<sup>41</sup> Ca	2.44e-7
<sup>54</sup> Mn	1.59e-7
<sup>55</sup> Fe	2.95e-5
<b>Total</b>	<b>3.43e-5</b>

## 2.5 REGULATORY COMPLIANCE

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

## 2.6 REFERENCES

- [2.0.1] American Concrete Institute, "Building Code Requirements for Structural Concrete", ACI 318-95, ACI, Detroit, Michigan.
- [2.0.2] American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures", ACI 349-85, ACI, Detroit, Michigan<sup>†</sup>
- [2.0.3] ~~Levy, et al., "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," Pacific Northwest Laboratory, PNL-6189, 1987.~~
- [2.0.4] NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," USNRC, Washington, D.C. Rev. 1 (1986).
- [2.0.5] J.W. McConnell, A.L. Ayers, and M.J. Tyacke, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Component According to Important to Safety," Idaho Engineering Laboratory, NUREG/CR-6407, INEL-95-0551, 1996.
- [2.0.6] NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities, March 2000
- [2.0.7] ASME Code, Section III, Subsection NF and Appendix F, and Code Section II, Part D, Materials, 1995, with Addenda through 1997.
- [2.0.8] *"Cladding Considerations for the Transportation and Storage of Spent Fuel," USNRC Interim Staff Guidance-11, Revision 3, November 17, 2003.*
- [2.0.9] *USNRC Memorandum from Christopher L. Brown to M. Wayne Hodges, "Scoping Calculations for Cladding Hoop Stresses in Low Burnup Fuel," dated January 29, 2004.*
- [2.1.1] ORNL/TM-10902, "Physical Characteristics of GE BWR Fuel Assemblies", by R.S. Moore and K.J. Notz, Martin Marietta (1989).
- [2.1.2] U.S. DOE SRC/CNEAF/96-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1994, Feb. 1996.
- [2.1.3] Deleted.

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<sup>†</sup> The 1997 edition of ACI-349 is specified for ISFSI pad and embedment design for deployment of the anchored HI-STORM 100A and HI-STORM 100SA.

- [2.1.4] Deleted.
- [2.1.5] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.
- [2.1.6] DOE Multi-Purpose Canister Subsystem Design Procurement Specification.
- [2.1.7] S.E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks", SAND-89-0018, Sandia National Laboratory, Oct., 1989.
- [2.1.8] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.
- [2.2.1] ASME Boiler & Pressure Vessel Code, American Society of Mechanical Engineers, 1995 with Addenda through 1997.
- [2.2.2] ASCE 7-88 (formerly ANSI A58.1), "Minimum Design Loads for Buildings and Other Structures", American Society of Civil Engineers, New York, NY, 1990.
- [2.2.3] ANSI N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More", June 1993.
- [2.2.4] Holtec Report HI-2012610, "Final Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 72-1008, latest revision.
- [2.2.5] Holtec Report HI-951251, "Safety Analysis Report for the HI-STAR 100 Cask System", NRC Docket No. 71-9261, latest revision.
- [2.2.6] "Debris Collection System for Boiling Water Reactor Consolidation Equipment", EPRI Project 3100-02 and ESEERCO Project EP91-29, October 1995.
- [2.2.7] Design Basis Tornado for Nuclear Power Plants, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, April 1974.
- [2.2.8] ANSI/ANS 57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (dry type)", American Nuclear Society, LaGrange Park, Illinois.
- [2.2.9] NUREG-0800, SRP 3.5.1.4, USNRC, Washington, DC.

- [2.2.10] United States Nuclear Regulatory Commission Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", August 1973 and Rev. 1, April 1976.
- [2.2.11] "Estimate of Tsunami Effect at Diablo Canyon Nuclear Generating Station, California." B.W. Wilson, PG&E (September 1985, Revision 1).
- [2.2.12] Deleted.
- [2.2.13] ~~Cunningham et al., "Evaluation of Expected Behavior of LWR Stainless Clad Fuel in Long Term Dry Storage", EPRI TR-106440, April 1996.Deleted.~~
- [2.2.14] ~~M.W. Schwartz and M.C. Witte, Lawrence Livermore National Laboratory, "Spent Fuel Cladding Integrity During Dry Storage", UCID-21181, September 1987.Deleted.~~
- [2.2.15] ~~PNL-4835, "Technical Basis for Storage of Zircaloy Clad Spent Fuel in Inert Gases", A.B. Johnson and E.R. Gilbert, Pacific Northwest Laboratories, September 1983.Deleted~~
- [2.2.16] Crane Manufacturer's Association of America (CMAA), Specification #70, 1988, Section 3.3.

## APPENDIX 2.A

### GENERAL DESIGN AND CONSTRUCTION REQUIREMENTS FOR THE ISFSI PAD FOR HI-STORM 100A

#### 2.A.1 General Comments

As stated in Section 2.0.4, an ISFSI slab that anchors a spent fuel storage cask should be classified as "important to safety." This classification of the slab follows from the provisions of 10CFR72, which require that the cask system retain its capacity to store spent nuclear fuel in a safe configuration subsequent to a seismic or other environmental event. Since the slab for anchored HI-STORM deployment is designated as ITS, the licensee is required to determine whether the reactor site parameters, including earthquake intensity and large missiles, are enveloped by the cask design bases. The intent of the regulatory criteria is to ensure that the slab meets all interface requirements of the cask design and the geotechnical characteristics of the ISFSI site.

This appendix provides general requirements for design and construction of the ISFSI concrete pad as an ITS structure, and also establishes the framework for ensuring that the ISFSI design bases are clearly articulated. The detailed design of the ISFSI pad for anchored HI-STORM deployment shall comply with the technical provisions set forth in this appendix.

#### 2.A.2 General Requirements for ISFSI Pad

1. Consistent with the provisions of NUREG-1567 [2.0.6], all concrete work shall comply with the requirements of ACI-349-97 [2.0.2].
2. All reinforcing steel shall be manufactured from high strength billet steel conforming to ASTM designation A615 Grade 60.
3. The ISFSI owner shall develop appropriate mixing, pouring, reinforcing steel placement, curing, testing, and documentation procedures to ensure that all provisions of ACI 349-97 [2.0.2] are met.
4. The placement, depth, and design and construction of the slab shall take into account the depth of the frost line at the ISFSI location. The casks transmit a very small amount of heat into the cask pad through conduction. The American Concrete Institute guidelines on reinforced concrete design of ground level slabs to minimize thermal and shrinkage induced cracking shall be followed.

5. General Requirements for Steel Embedment: The steel embedment, excluding the pre-tensioned anchorage studs, is required to follow the provisions stipulated in ACI 349-97 [2.0.2], Appendix B "Steel Embedment" and the associated Commentary on Appendix B, as applicable. Later editions of this Code may be used provided a written reconciliation is performed. An example of one acceptable embedment configuration is provided in Figure 2.A.1. Site-specific embedment designs may vary from this example, depending on the geotechnical characteristics of the site-specific foundation. The embedment designer shall consider any current, relevant test data in designing the pad embedment for HI-STORM 100A and HI-STORM 100SA.
6. The ISFSI owner shall ensure that pad design analyses, using interface loads provided in this report, demonstrate that all structural requirements of NUREG-1567 and ACI-349-97 are satisfied.
7. Unless the load handling device is designed in accordance with ANSI N14.6 and incorporates redundant drop protection features, the ISFSI owner shall ensure that a permissible cask carry height is computed for the site-specific pad/foundation configuration such that the design basis deceleration set forth in this FSAR are not exceeded in the event of a handling accident involving a vertical drop.
8. The ISFSI owner shall ensure that the pad/foundation configuration provides sufficient safety margins for overall kinematic stability of the cask/pad/foundation assemblage.
9. The ISFSI owner shall ensure that the site-specific seismic inputs, established at the top surface of the ISFSI pad, are bounded by the seismic inputs used as the design basis for the attachment components. If required, the ISFSI owner shall perform additional analyses to ensure that the site-specific seismic event or durations greater than the design basis event duration analyzed in this report, do not produce a system response leading to structural safety factors (defined as allowable stress (load) divided by calculated stress (load)) less than 1.0. Table 2.0.5 and Table 2.2.8 provide the limiting values of ZPAs in the three orthogonal directions that must not be exceeded at an ISFSI site (on the pad top surface) to comply with the general CoC for the HI-STORM 100A (and 100SA) System.
10. An ISFSI pad used to support anchored HI-STORM overpacks, unlike the case of free standing overpacks, may experience tensile (vertically upward) anchorage forces in addition to compression loads. The reinforcing steel (pattern and quantity) must be selected to meet the demands of the anchorage forces under seismic and other environmental conditions that involve destabilizing loadings (such as the large tornado missile defined in this FSAR).

### 2.A.3 Steel Embedment for Anchored Casks

Figure 2.A.1 shows a typical fastening arrangement for the HI-STORM 100A System. The details of the rebars in the pad (which are influenced by the geotechnical characteristics of the foundation and its connection to the underlying continuum) are not shown in Figure 2.A.1. Representative dimensions of the embedment and anchorage system are provided in Table 2.A.1.

The embedment detail illustrated in Figure 2.A.1 is designed to resist a load equal to the ASME Code, Section III Appendix F Level D load capacity of the cask anchor studs. The figure does not show the additional reinforcement required to ensure that tensile cracking of concrete is inhibited (see Figure B-4 in the Commentary ACI-349R-97) as this depends on the depth chosen for the ITS ISFSI pad concrete. The ACI Code contemplates ductile failure of the embedment steel and requires that the ultimate load capacity of the steel embedment be less than the limit pullout strength of the concrete surrounding the embedment that resists the load transferred from the cask anchor stud. If this criterion cannot be assured, then additional reinforcement must be added to inhibit concrete cracking (per Subsection B.4.4 of Appendix B of ACI-349-97).

The anchor stud receptacle described in Figure 2.A.1 is configured so that the cask anchor studs (which interface with the overpack baseplate as well as the pad embedment per Table 2.0.5 and are designed in accordance with ASME Section III, Subsection NF stress limits), sits flush with the ISFSI top surface while the cask is being positioned. Thus, a translocation device such as an “air pad” (that requires a flat surface) can be used to position the HI-STORM overpack at the designated location. Subsequent to positioning of the cask, the cask anchor stud is raised, the anchor stud nut installed, and the anchor stud preload applied. The transfer of load from the cask anchor stud to the embedment is through the bearing surface of the lower head of the cask anchor stud and the upper part of the anchor stud receptacle shown in the figure. The members of the anchoring system illustrated in Figure 2.A.1, as well as other geometries developed by the ISFSI designer, must meet the following criteria:

- i. The weakest structural link in the system shall be in the ductile member. In other words, the tension capacity of the anchor stud/anchor receptacle group (based on the material ultimate strengths) shall be less than the concrete pull-out strength (computed with due recognition of the rebars installed in the pad).
- ii. The maximum ratio of embedment plus cask anchor stud effective tensile stiffness to the effective compressive stiffness of the embedment plus concrete shall not exceed 0.25 in order to ensure the effectiveness of the pre-load.
- iii. The maximum axial stress in the cask anchor studs under normal and seismic conditions shall be governed by the provisions of ASME Section III Subsection NF (1995).
- iv. The load-bearing members of the HI-STORM 100A anchorage system shall be considered important-to-safety. This includes the following components shown in Figure 2.A.1: anchor stud and nut, top ring, upper collar, anchor receptacle, and anchor ring.



For sites with lower ZPA DBE events, compared to the limiting ZPAs set down in this FSAR, the size of the anchor studs and their number can be appropriately reduced. However, the above three criteria must be satisfied in all cases.

Table 2.A.1

Typical Embedment and Anchoring Data\*

Nominal diameter of the anchor stud, (inch)	2
Thickness of the embedment ring, (inch)	2
I.D. of the embedment ring, (inch)	130
Anchor receptacle: Upper Position O.D. and I.D. (inch) Lower portion O.D. and I.D. (inch)	O.D.: 2.5 / I.D.: 2.125 (min.) O.D.: 4.875 / I.D.: 3.625 (min.)
Depth of anchor receptacle collar, d, (inch)	2.5
Free fall height of the anchor stud, $h_e$ , (inch)	8
<b>Representative Materials of Construction are as follows:<sup>†</sup></b>	
Anchor Studs:	Per Table 2.0.4
Anchor Receptacle:	Low carbon steel such as A-36, A-105
Top Ring, Upper Collar, Anchor Ring:	Low carbon steel such as A-36, A-516-Gr. 70

\* Refer to Figure 2.A.1

<sup>†</sup> The ISFSI designer shall ensure that all permanently affixed embedment parts (such as the anchor receptacle) made from materials vulnerable to deleterious environmental effects (e.g. low carbon steel) are protected through the use of suitably engineered corrosion barrier. Alternatively, the selected material of construction must be innately capable of withstanding the long term environmental conditions at the ISFSI site.

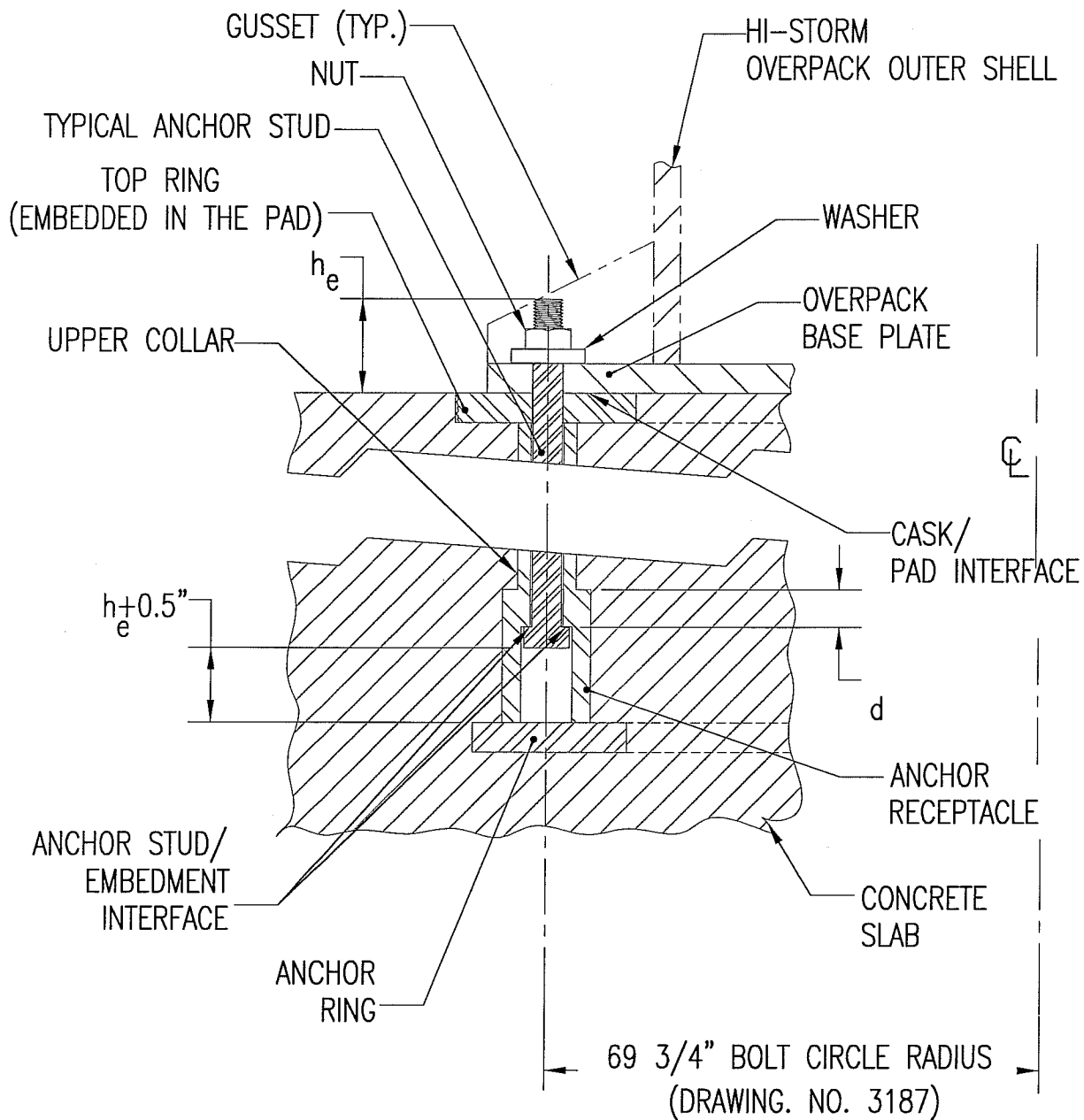


Figure 2.A.1;  
Typical HI-STORM/ISFSI pad Fastening Detail

Note: Rebars in the ISFSI pad and sub-surface soil/rock continuum not shown.

## Appendix 2.B The Forced Helium Dehydration (FHD) System

### 2.B.1 System Overview

The Forced Helium Dehydration (FHD) system is used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump *or an inert gas* has been expelled in the water blowdown operation. The FHD system is required to be used for MPCs containing at least one high burnup fuel assembly. *The FHD method of moisture removal* and is optional for *all other* MPCs. ~~containing all moderate burnup fuel assemblies.~~

Expelling the water from the MPC using a conventional pump *or a water displacement method using inert gas* would remove practically all of the contained water except for the small quantity remaining on the MPC baseplate below the bottom of the drain line and an even smaller adherent amount wetting the internal surfaces. A skid-mounted, closed loop dehydration system will be used to remove the residual water from the MPC such that the partial pressure of the trace quantity of water vapor in the MPC cavity gas is brought down to  $\leq 3$  torr. The FHD system, engineered for this purpose, shall utilize helium gas as the working substance.

The FHD system, schematically illustrated in Figure 2.B.1, can be viewed as an assemblage of four thermal modules, namely, (i) the condensing module, (ii) the demister module, (iii) the helium circulator module and (iv) the pre-heater module. The condensing module serves to cool the helium/vapor mixture exiting the MPC to a temperature well below its dew point such that water may be extracted from the helium stream. The condensing module is equipped with suitable instrumentation to provide a direct assessment of the extent of condensation that takes place in the module during the operation of the FHD system. The demister module, engineered to receive partially cooled helium exiting the condensing module, progressively chills the recirculating helium gas to a temperature that is well below the temperature corresponding to the partial pressure of water vapor at 3 torr.

The motive energy to circulate helium is provided by the helium circulator module, which is sized to provide the pressure rise necessary to circulate helium at the requisite rate. The last item, labeled the pre-heater module, serves to pre-heat the flowing helium to the desired temperature such that it is sufficiently warm to boil off any water present in the MPC cavity.

The pre-heater module, in essence, serves to add supplemental heat energy to the helium gas (in addition to the heat generated by the stored SNF in the MPC) so as to facilitate rapid conversion of water into vapor form. The heat input from the pre-heater module can be adjusted in the manner of a conventional electric heater so that the recirculating helium entering the MPC is sufficiently dry and hot to evaporate water, but not unduly hot to place unnecessary thermal burden on the condensing module.

The FHD system described in the foregoing performs its intended function by continuously removing water entrained in the MPC through successive cooling, moisture removal and reheating of the working substance in a closed loop. In a classical system of the FHD genre, the moisture removal operation occurs in two discrete phases. In the beginning of the FHD system's

operation (Phase 1), the helium exiting the MPC is laden with water vapor produced by boiling of the entrained bulk water. The condensing module serves as the principal device to condense out the water vapor from the helium stream in Phase 1. Phase 1 ends when all of the bulk water in the MPC cavity is vaporized. At this point, the operation of the FHD system moves on to steadily lowering the relative humidity and bulk temperature of the circulating helium gas (Phase 2). The demoisturizer module, equipped with the facility to chill flowing helium, plays the principal role in the dehydration process in Phase 2.

## 2.B.2 Design Criteria

The design criteria set forth below are intended to ensure that design and operation of the FHD system will drive the partial pressure of the residual vapor in the MPC cavity to  $\leq 3$  torr if the ~~gas temperature of helium exiting the demoisturizer~~ *specified temperature or dew point* value and duration criteria. ~~provided in the HI-STORM technical specifications.~~ The FHD system shall be designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

- i. The temperature of helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.
- ii. The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).
- iii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulated flow regime in the MPC cavity.
- iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr. *The limit will be met if the helium gas temperature at the demoisturer outlet is verified by measurement to remain  $\leq 21^\circ\text{F}$  for a period of  $\geq 30$  minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain  $\leq 22.9^\circ\text{F}$  for  $\geq 30$  minutes.*

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

- i. The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point of 120°F or less.
- ii. The demoisturizer module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in Section 2.B.1) has been completed.
- iii. The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

- iv. The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

### 2.B.3 Analysis Requirements

The design of the FHD system shall be subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this FSAR.

- i. System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.
- ii. System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demister modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach  $\leq 3$  torr if the temperature of the gas exiting the demister is predicted to be at a maximum of 21°F for 30 minutes.
- iii. Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

### 2.B.4 Acceptance Testing

The first FHD system designed and built for the MPC drying function required by HI-STORM's technical specifications shall be subject to confirmatory testing as follows:

- a. A representative quantity of water shall be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.
- b. The MPC cavity drying test shall be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).
- c. The drain and vent line RVOAs on the MPC lid shall be connected to the terminals located in the pre-heater and condensing modules of the FHD system, respectively.

- d. The FHD system shall be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demister module has been at or below 21°F for thirty minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

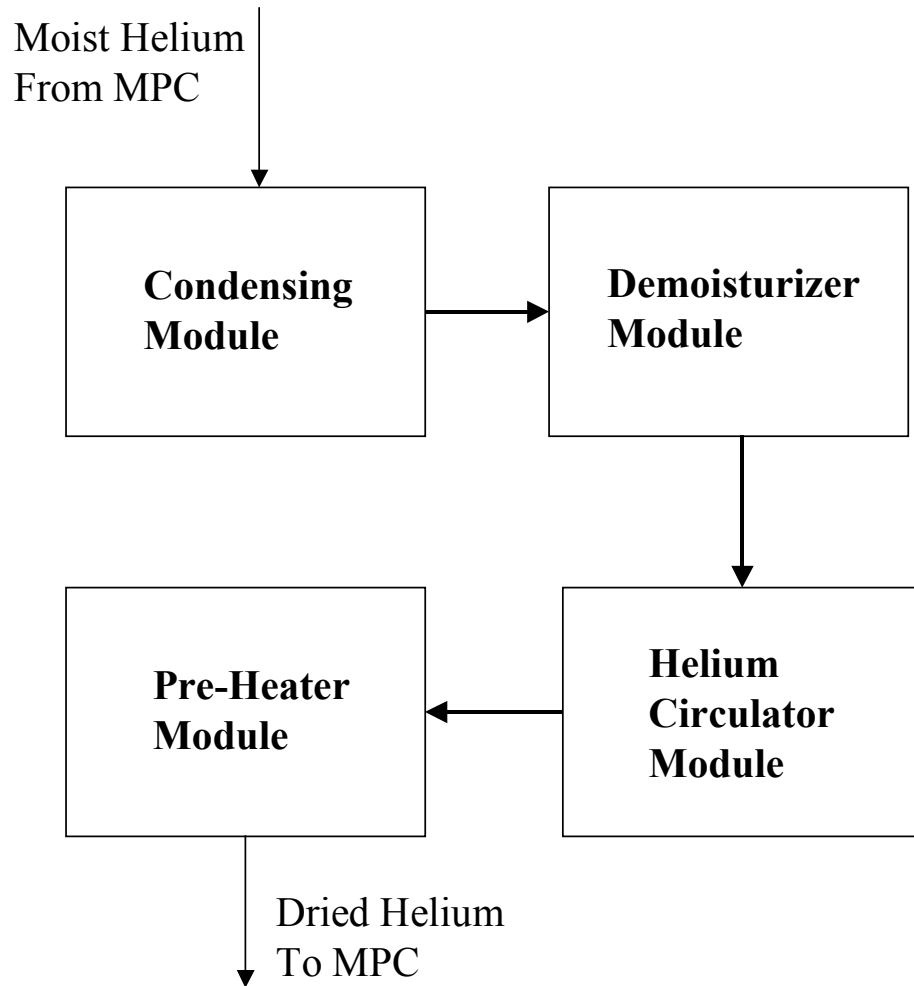


FIGURE 2.B.1: SCHEMATIC OF THE FORCED HELIUM DEHYDRATION SYSTEM



## Appendix 2.C

### The Supplemental Cooling System

#### 2.C.1 Purpose

The Supplemental Cooling System (SCS) will be utilized, as necessary, to maintain the peak fuel cladding temperature below the limit set forth in Chapter 2 of the FSAR during normal short-term the MPC onsite transport operations.

#### 2.C.2 General Description and Requirements

The SCS is a water circulation system for cooling the MPC inside the HI-TRAC transfer cask during on-site transport. The system consists of a skid-mounted coolant pump and an air-cooled heat exchanger. During normal SCS operation, heat is removed by water from the HI-TRAC annulus and rejected to the heat sink (ambient air) across the air cooler heat exchange surfaces. The SCS shall be designed to meet the following criteria:

- (i) The pump is sized to limit the coolant temperature rise (from annulus inlet to outlet) to a reasonably low value (20°F) and the air-cooled heat exchanger sized for the design basis heat load at an ambient air temperature of 100°F. The pump and air-cooler fan are powered by electric motors with a backup power supply for uninterrupted operation.
- (ii) The closed loop cooling circuit will utilize a contamination-free fluid medium in contact with the external surfaces of the MPC and inside surfaces of the HI -TRAC transfer cask to minimize corrosion. Figure 2.C.1 shows a typical P&ID for a SCS.
- (iii) The number of active components in the SCS will be minimized.
- (iv) All passive components such as tubular heat exchangers, manually operated valves and fittings shall be designed to applicable standards (TEMA, ANSI).

#### 2.C.3 Thermal/Hydraulic Design Criteria

- (i) The heat dissipation capacity of the SCS shall be equal to or greater than the minimum necessary to ensure that the peak cladding temperature is below the ISG-11, Rev. 3 limit of 400°C (752°F). All heat transfer surfaces in heat exchangers shall be assumed to be fouled to the maximum limits specified in a widely used heat exchange equipment standard such as the Standards of Tubular Exchanger Manufacturers Association.
- (ii) The coolant utilized to extract heat from the MPC shall be high purity water. Anti-freeze may be used to prevent water from freezing if warranted by operating conditions.

#### 2.C.4 Mechanical Requirements

- (i) *All pressure boundaries (as defined in the ASME Boiler and Pressure Vessel Code, Section VIII Division 1) shall have pressure ratings that are greater than the maximum system operating pressure by at least 15 psi.*
- (ii) *All ASME Code components shall comply with Section VIII Division 1 of the ASME Boiler and Pressure Vessel Code.*
- (iii) *Prohibited Materials*

*The following materials will not be in contact with the system coolant in the SCS.*

- *Lead*
  - *Mercury*
  - *Sulfur*
  - *Saran*
  - *Silastic L8-53*
  - *Cadmium*
  - *Tin*
  - *Antimony*
  - *Bismuth*
  - *Mischmetal*
  - *Neoprene or similar gasket materials made of halogen containing elastomers*
  - *Phosphorus*
  - *Zinc*
  - *Copper and Copper Alloys*
  - *Rubber-bonded asbestos*
  - *Nylon*
  - *Magnesium oxide (e.g., insulation)*
  - *Materials that contain halogens in amounts exceeding 75 ppm*
- (iv) *All gasketed and packed joints shall have a minimum design pressure rating of the pump shut-off pressure plus 15 psi.*
  - (v) *The SCS skid shall be equipped with appropriate lifting lugs to permit its handling by the plant's lifting devices in full compliance with NUREG-0612 provisions.*

#### 2.C.5 Regulatory Requirements

*The SCS is classified as not important-to-safety.*

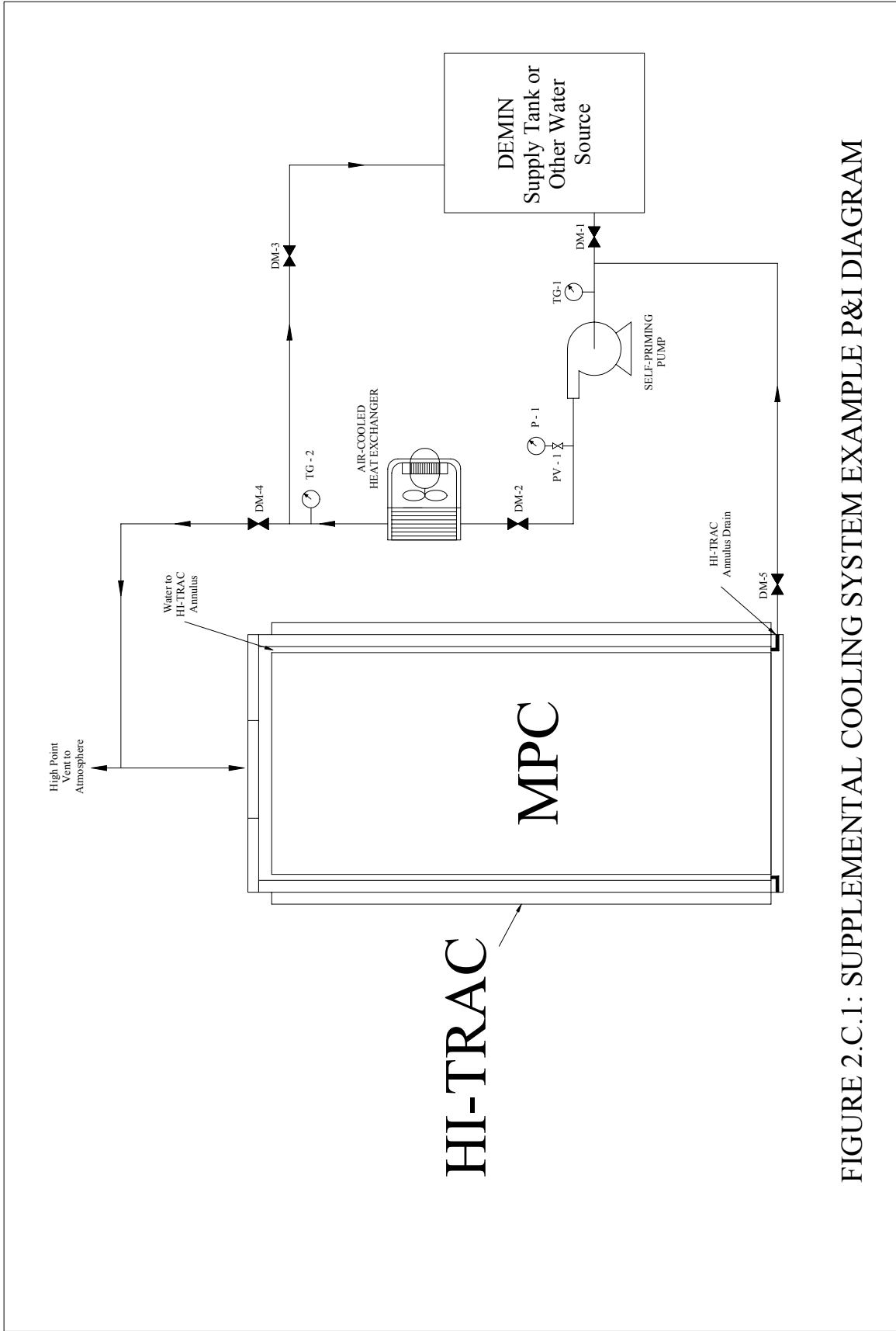


FIGURE 2.C.1: SUPPLEMENTAL COOLING SYSTEM EXAMPLE P&I DIAGRAM

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

In this chapter, the structural components of the HI-STORM 100 System that are important to safety (ITS) are identified and described. The objective of the structural analyses is to ensure that the integrity of the HI-STORM 100 System is maintained under all credible loads for normal, off-normal, and design basis accident/natural phenomena. The chapter results support the conclusion that the confinement, criticality control, radiation shielding, and retrievability criteria set forth by 10CFR72.236(l), 10CFR72.124(a), 10CFR72.104, 10CFR72.106, and 10CFR72.122(l) are met. In particular, the design basis information contained in the previous two chapters and in this chapter provides sufficient data to permit structural evaluations to demonstrate compliance with the requirements of 10CFR72.24. To facilitate regulatory review, the assumptions and conservatism's inherent in the analyses are identified along with a complete description of the analytical methods, models, and acceptance criteria. A summary of other material considerations, such as corrosion and material fracture toughness is also provided. Design calculations for the HI-TRAC transfer cask are included where appropriate to comply with the guidelines of NUREG-1536.

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

This revision to the HI-STORM Safety Analysis Report, the first since the HI-STORM 100 System was issued a Part 72 Certificate-of-Compliance, incorporates several features into the structural analysis to respond to the changing needs of the U.S. nuclear power generation industry. The most significant changes to this chapter for this revision are:

- The incorporation of structural results associated with the MPC-32 and the MPC-24E/24EF fuel baskets. In the case of the MPC-32, this revision simply returns results of analyses that were contained in this chapter prior to the initial CoC. In the case of the 24E basket, the new results are based on the same structural analysis model used for all the other baskets evaluated.
- The revision of the analyses of free thermal expansion and MPC canister shell to incorporate the changed temperature distribution from the inclusion of the thermosiphon effect (convective heat transfer inside the canister).
- The introduction of new analyses that permit the use of additional damaged fuel canisters in the HI-STORM 100.

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

- The inclusion of a short version of the HI-STORM overpack (designated as HI-STORM 100S) to accommodate plants with reduced clearances. In general, we show that the HI-STORM 100S is bounded by results previously obtained.
- Revisions to approved HI-TRAC analyses to accommodate fabrication enhancements.
- Enhancement of the handling accident and tipover analyses to provide an additional qualified reference ISFSI pad configuration with higher strength concrete.
- Introduction of an anchored HI-STORM (designated as HI-STORM 100A). This enhancement permits use of a HI-STORM at sites in high seismic zones where a free standing cask is not acceptable.

The organization of technical information in this chapter follows the format and content guidelines of USNRC Regulatory Guide 3.61 (February 1989). The FSAR ensures that the responses to the review requirements listed in NUREG-1536 (January 1997) are complete and comprehensive. The areas of NRC staff technical inquiries, with respect to structural evaluation in NUREG-1536, span a wide array of technical topics within and beyond the material in this chapter. To facilitate the staff's review to ascertain compliance with the stipulations of NUREG-1536, Table 3.0.1 "Matrix of NUREG-1536 Compliance - Structural Evaluation", is included in this chapter. A comprehensive cross-reference of the topical areas set forth in NUREG-1536, and the location of the required compliance information is contained in Table 3.0.1.

Section 3.7 describes in detail HI-STORM 100 System's compliance to NUREG-1536 Structural Evaluation Requirements.

The HI-STORM 100 System matrix of compliance table given in this section is developed with the supposition that the storage overpack is designated as a steel structure that falls within the purview of subsection 3.V.3 "Other Systems Components Important to Safety" (page 3-28 of NUREG-1536), and therefore, does not compel the use of reinforced concrete. (Please refer to Table 1.0.3 for an explicit statement of exception on this matter). The concrete mass installed in the HI-STORM 100 overpack is accordingly equipped with "plain concrete" for which the sole applicable industry code is ACI 318.1 (92). Plain concrete, in contrast to reinforced concrete, is the preferred shielding material HI-STORM 100 because of three key considerations:

- (i) Plain concrete is more amenable to a void free pour than reinforced concrete in narrow annular spaces typical of ventilated vertical storage casks.
- (ii) The tensile strength bearing capacity of reinforced concrete is not required to buttress the steel weldment of the HI-STORM 100 overpack.

- (iii) The compression and bearing strength capacity of plain concrete is unaffected by the absence of rebars. A penalty factor, on the compression strength, pursuant to the provisions of ACI-318.1 is, nevertheless, applied to insure conservatism. However, while plain concrete is the chosen shielding embodiment for the HI-STORM 100 storage overpack, all necessary technical, procedural Q.C., and Q.A. provisions to insure nuclear grade quality will be implemented by utilizing the relevant sections from ACI-349 (85) as specified in Appendix 1.D.

In other words, guidelines of NUREG 1536 pertaining to reinforced concrete are considered to insure that the material specification, construction quality control and quality assurance of the shielding concrete comply with the provisions of ACI 349 (85). These specific compliance items are listed in the compliance matrix.

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

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
IV.1.a	ASME B&PV Compliance		
	NB	3.1.1	Tables 2.2.6,2.2.7
	NG	3.1.1	Tables 2.2.6,2.2.7
IV.2	Concrete Material Specification		Appendix 1.D
IV.4	Lifting Devices	3.1; 3.4; <del>3.D</del> ; <del>3.E</del> ; <del>3.AC</del>	
V.	Identification of SSC that are ITS		Table 2.2.6
“	Applicable Codes/Standards	3.6.1	Table 2.2.6
“	Loads		Table 2.2.13
“	Load Combinations	3.1.2.1.2; Tables 3.1.1-3.1.5	Table 2.2.14
“	Summary of Safety Factors	3.4.3; 3.4.4.2; 3.4.4.3.1-3.4.6-3.4.9; Tables 3.4.3-3.4.9	
“	Design/Analysis Procedures	Chapter 3 <del>plus Appendices</del>	
“	Structural Acceptance Criteria		Tables 2.2.10-2.2.12

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Material/QC/Fabrication	Table 3.4.2	Chap. 9; Chap. 13
“	Testing/In-Service Surveillance		Chap. 9; Chap. 12
“	Conditions for Use		Table 1.2.6; Chaps. 8,9,12
V.1.a	Description of SSC	3.1.1	1.2
V.1.b.i.(2)	Identification of Codes & Standards		Tables 2.2.6, 2.2.7
V.1.b.ii	Drawings/Figures		1.5
“	Identification of Confinement Boundary		1.5; 2.3.2; 7.1; Table 7.1.1
“	Boundary Weld Specifications	3.3.1.4	1.5; Table 7.1.2
“	Boundary Bolt Torque	NA	
“	Weights and C.G. Location	Tables 3.2.1-3.2.4	
“	Chemical/Galvanic Reactions	3.4.1; Table 3.4.2	
V.1.c	Material Properties	3.3; Tables 3.3.1-3.3.5	1.A; 1.C; 1.D
“	Allowable Strengths	Tables 3.1.6-3.1.17	Tables 2.2.10-2.2.12; 1.D
“	Suitability of Materials	3.3; Table 3.4.2	1.A; 1.B; 1.D
“	Corrosion	3.3	
“	Material Examination before Fabrication		9.1.1



**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Material Testing and Analysis		9.1; Table 9.1.1;1.D
“	Material Traceability		9.1.1
“	Material Long Term Performance	3.3; 3.4.11; 3.4.12	9.2
“	Materials Appropriate to Load Conditions		Chap. 1
“	Restrictions on Use		Chap. 12
“	Temperature Limits	Table 3.1.17	Table 2.2.3
“	Creep/Slump	3.4.4.3.3.2; 3.F	
“	Brittle Fracture Considerations	3.1.2.3; Table 3.1.18	
“	Low Temperature Handling		2.2.1.2
V.1.d.i.(1)	Normal Load Conditions		2.2.1; Tables 2.2.13,2.2.14
“	Fatigue	3.1.2.4	
“	Internal Pressures/Temperatures for Hot and Cold Conditions	3.4.4.1	2.2.2; Tables 2.2.1,2.2.3
“	Required Evaluations		
“	Weight+Pressure	3.4.4.3.1.2	
“	Weight/Pressure/Temp.	3.4.4.3.1.2	
“	Free Thermal Expansion	3.4.4.2; 3.U; 3.V; 3.W; 3.I; 3.AF; 3.AQ	4.4.5; Figure 4.4.30 Tables 4.4.15, 4.5.4

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
V.1.d.i.(2)	Off-Normal Conditions		2.2.2; Tables 2.2.13, 2.2.14; 11.1
V.1.d.i.(3)	Accident Level Events and Conditions	Tables 3.1.1, 3.1.2	2.2.3; Tables 2.2.13, 2.2.14; 11.2
V.1.d.i.(3).(a)	Storage Cask Vertical Drop	3.1.2.1.1.2; 3.4.10; 3.A	2.2.3.1
“	Storage Cask Tipover	3.1.2.1.1.1; 3.4.10; 3.A	2.2.3.2
“	Transfer Cask Horizontal Drop	3.4.9; 3.Z; 3.AL; 3.AN	2.2.3.1
V.1.d.i.(3).(b)	Explosive Overpressure	3.1.2.1.1.4; 3.AK	2.2.3.10
V.1.d.i.(3).(c)	Fire		
“	Structural Evaluations	3.4.4.2	2.2.3.3
“	Material Properties		11.2
“	Material Suitability	3.1.2.2; 3.3.1.1	Table 2.2.3; 11.2
V.1.d.i.(3).(d)	Flood		
“	Identification	3.1.2.1.1.3; 3.4.6	2.2.3.6
“	Cask Tipover	3.4.6	
“	Cask Sliding	3.4.6	
“	Hydrostatic Loading	3.1.2.1.1.3; 3.4.6	72-1008(3.H)
“	Consequences		11.2
V.1.d.i.(3).(e)	Tornado Winds		
“	Specification	3.1.2.1.1.5	2.2.3.5; Table 2.2.4
“	Drag Coefficients	3.4.8; 3.C	
“	Load Combination	3.4.8; 3.C	

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION <sup>†</sup>**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Overturning –Transfer	NA	
V.1.d.i.(3).(f)	Tornado Missiles		
“	Missile Parameters	3.1.2.1.1.5	Table 2.2.5
“	Tipover	3.4.8; <del>3.C</del>	
“	Damage	3.4.8.1; 3.4.8.23.B; 3.G; 3.H; 3.Z; 3.AM	
“	Consequences	3.4.8.1; 3.4.8.2	11.2
V.1.d.i.(3).(g)	Earthquakes		
“	Definition of DBE	3.1.2.1.1.6; 3.4.7	2.2.3.7; Table 2.2.8
“	Sliding	3.4.7	
“	Overturning	3.4.7	
“	Structural Evaluations	3.4.7; <del>3B</del>	11.2
V.1.d.i.(4).(a)	Lifting Analyses		
“	Trunnions		
“	Requirements	3.1.2.1.2; 3.4.3.1; 3.4.3.2	72-1008(3.4.3); 2.2.1.2
“	Analyses	3.4.3.1; 3.4.3.2; <del>3.D; 3.E; 3.AC; 3.AE</del>	72-1008(3.4.3)
“	Other Lift Analyses	3.4.3.7-3.4.3.9; <del>3.D; 3.AB; 3.AC; 3.AE; 3.AD; 3.AI; 3.AJ</del>	
V.1.d.i.(4).(b)	Fuel Basket		
“	Requirements	3.1.2.1.2; Table 3.1.3	
“	Specific Analyses	3.4.4.2; 3.4.4.3; 3.6.3; <del>3.U; 3.W; 3.I; 3.Y</del>	72-1008(3.4.4.3.1.2; 3.4.4.3.1.6; <del>3.AA; 3.M; 3.H; 3.I</del> )

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Dynamic Amplifiers	<del>3.4.4.4.13-X</del>	
“	Stability	3.4.4.3; 3.4.4.4; <del>3-AK</del>	72-1008(Figures 3.4.27-32)
V.1.d.i.(4).(c)	Confinement Closure Lid Bolts		
“	Pre-Torque	NA	
“	Analyses	NA	
“	Engagement Length	NA	
“	Miscellaneous Bolting		
“	Pre-Torque	<del>3.4.3.7; 3.4.3.83-AC</del>	
“	Analyses	<del>3.4.4.3.2.23-L</del>	
“	Engagement Length	3.4.3.5; 3.4.3.7; <del>3.4.3.83-AC; 3-D</del>	
V.1.d.i.(4)	Confinement		
“	Requirements	3.1.2.1.2; Table 3.1.4	Chap. 7
“	Specific Analyses	3.6.3; Tables 3.4.3, 3.4.4; <del>3-D</del>	72-1008(3.E; 3.K; 3.I; <del>3-AA-3.4.4.3.1.5</del> )
“	Dynamic Amplifiers	<del>3-X</del> ; 3.4.4.1	
“	Stability	3.4.4.3.1	72-1008(3.H)
“	Overpack		
“	Requirements	3.1.2.1.2; Tables 3.1.1, 3.1.5	
“	Specific Analyses	3.6.3; <del>3-B; 3-D; 3-L; 3-M;</del> <del>3-AC; 3-D; 3.4.4.3; 3-K;</del> <del>3-AK; 3-AR; 3-AS</del>	

**TABLE 3.0.1 (CONTINUED)**  
**MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †**

<b>PARAGRAPH IN NUREG-1536</b>	<b>NUREG-1536 COMPLIANCE ITEM</b>	<b>LOCATION IN FSAR CHAPTER 3</b>	<b>LOCATION OUTSIDE OF FSAR CHAPTER 3</b>
“	Dynamic Amplifiers	3.4.4.3.2; <del>3.X</del>	
“	Stability	3.4.4.3; Table 3.1.1; 3.4.4.5; <del>3.AK</del>	
“	Transfer Cask		
“	Requirements	3.1.2.1.2; Table 3.1.5	
“	Specific Analyses	3.4.4.3; 3.6.3; <del>3.E; 3.H; 3.I;</del> <del>3.Z; 3.AD; 3.AE; 3.AA;</del> <del>3.AI; 3.AB; 3.AD; 3.AG;</del> <del>3.F; 3.AH; 3.AJ; 3.AL;</del> <del>3.AM</del>	
“	Dynamic Amplifiers	<del>3.4.4.4.13.X</del>	
“	Stability	NA	2.2.3.1

† Legend for Table 3.0.1

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

72-1008                      HI-STAR 100 Docket Number where the referenced item is located  
 NA                              Not Applicable for this item

~~Appendices 3.N-3.T have been relocated to the Calculation Package, HI 2002481 as of this revision.~~

## 3.1 STRUCTURAL DESIGN

### 3.1.1 Discussion

The HI-STORM 100 System consists of three principal components: the Multi-Purpose Canister (MPC), the storage overpack, and the transfer cask. The MPC is a hermetically sealed, welded structure of cylindrical profile with flat ends and a honeycomb fuel basket. A complete description is provided in Subsection 1.2.1.1 wherein the anatomy of the MPC and its fabrication details are presented with the aid of figures. The MPCs utilized in the HI-STORM 100 System are identical to those for the HI-STAR 100 System submitted under Dockets 72-1008 and 71-9261. The evaluation of the MPCs presented herein draws upon the work described in those earlier submittals. In this section, the discussion is confined to characterizing and establishing the structural features of the MPC, the storage overpack, and the HI-TRAC transfer cask. Since a detailed discussion of the HI-STORM 100 Overpack and HI-TRAC transfer cask geometries is presented in Section 1.2, attention is focused here on structural capabilities and their inherent margins of safety for housing the MPC. Detailed design drawings for the HI-STORM 100 System are provided in Section 1.5.

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

- **Ability to Dissipate Heat:** The thermal energy produced by the stored spent fuel must be transported to the outside surface of the MPC such that the prescribed temperature limits for the fuel cladding and for the fuel basket metal walls are not exceeded.
- **Ability to Withstand Large Impact Loads:** The MPC, with its payload of nuclear fuel, must be sufficiently robust to withstand large impact loads associated with the postulated handling accident events. Furthermore, the strength of the MPC must be sufficiently isotropic to meet structural requirements under a variety of handling and tip-over accidents.
- **Restraint of Free End Expansion:** The membrane and bending stresses produced by restraint of free-end expansion of the fuel basket are categorized as primary stresses. In view of the concentration of heat generation in the fuel basket, it is necessary to ensure that structural constraints to its external expansion do not exist.

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

From the description presented in Chapter 1, the MPC enclosure vessel is the confinement vessel designed to meet ASME Code, Section III, Subsection NB stress limits. The enveloping canister shell, the baseplate, and the lid system form a complete confinement boundary for the stored fuel that is referred to as the "enclosure vessel". Within this cylindrical shell confinement vessel is an integrally welded assemblage of cells of square cross sectional openings for fuel storage, referred to herein as the fuel basket. The fuel basket is analyzed under the provisions of Subsection NG of Section III of the ASME Code. All multi-purpose canisters designed for deployment in the HI-STORM 100 and HI-STAR 100 systems are exactly alike in their external dimensions. The essential

difference between the MPCs lies in the fuel baskets. Each fuel storage MPC is designed to house fuel assemblies with different characteristics. Although all fuel baskets are configured to maximize structural ruggedness through extensive inter-cell connectivity, they are sufficiently dissimilar in structural details to warrant separate evaluations. Therefore, analyses for each of the MPC types were carried out to ensure structural compliance. Inasmuch as no new MPC designs are introduced in this application, and all MPC designs were previously reviewed by the USNRC under Docket 72-1008, the MPC analyses submitted under Docket Numbers 72-1008 and 71-9261 for the HI-STAR 100 System are not reproduced herein unless they need to be modified by HI-STORM 100 conditions or geometry differences. Analyses provided in the HI-STAR 100 System safety analysis reports that are applicable to the HI-STORM 100 System are referenced in this FSAR by docket number and subsection or appendix.

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

Some of the key structural functions of the MPC in the storage mode are:

1. To position the fuel in a subcritical configuration, and
2. To provide a confinement boundary.

Some of the key structural functions of the overpack in the storage mode are:

1. To serve as a missile barrier for the MPC,
2. To provide flow paths for natural convection,
3. To ensure stability of the HI-STORM 100 System, and
4. To maintain the position of the radiation shielding.
5. To allow movement of the overpack with a loaded MPC inside.

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

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

- The closure system for the MPCs consists of two components, namely, the MPC lid and the closure ring. The MPC lid can be either a single thick circular plate continuously welded to the MPC shell along its circumference or two dual lids welded around their common periphery. The MPC closure system is shown in the Design Drawings in Section 1.5. The MPC lid is equipped with vent and drain ports which are utilized for evacuating moisture and air from the MPC following fuel loading, and subsequent backfilling with an inert gas (helium) at a specified mass. The vent and drain ports are covered by a cover plate and welded before the closure ring is installed. The closure ring is a circular annular plate edge-welded to the MPC lid and shell. The two closure members are interconnected by welding around the inner diameter of the ring. Lift points for the MPC are provided in the MPC lid.
- The MPC fuel baskets consist of an array of interconnecting plates. The number of storage cells formed by this interconnection process varies depending on the type of fuel being stored. Basket designs containing cell configurations for PWR and BWR fuel have been designed and are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell. Welding of the basket plates along their edges essentially renders the fuel basket into a multiflange beam. Figure 3.1.1 provides an isometric illustration of a fuel basket for the MPC-68 design.
- The MPC basket is separated from its supports by a gap. The gap size decreases as a result of thermal expansion (depending on the magnitude of internal heat generation from the stored spent fuel). The provision of a small gap between the basket and the basket support structure is consistent with the natural thermal characteristics of the MPC. The planar temperature distribution across the basket, as shown in Section 4.4, approximates a shallow parabolic profile. This profile will create high thermal stresses unless structural constraints at the interface between the basket and the basket support structure are removed.
- The MPCs will be loaded with fuel with widely varying heat generation rates. The basket/basket support structure gap tends to be reduced for higher heat generation rates due to increased thermal expansion rates. Gaps between the fuel basket and the basket support structure are specified to be sufficiently large such that a gap exists around the periphery after any thermal expansion.
- A small number of flexible thermal conduction elements (thin aluminum tubes) are interposed between the basket and the MPC shell. The elements are designed to be resilient. They do not provide structural support for the basket, and thus their resistance to thermal growth is negligible.



It is quite evident from the geometry of the MPC that a critical loading event pertains to the drop condition when the MPC is postulated to undergo a handling side drop (the longitudinal axis of the MPC is horizontal) or tip-over. Under the side drop or tip-over condition the flat panels of the fuel basket are subject to an equivalent pressure loading that simulates the deceleration-magnified inertia load from the stored fuel and the MPC's own metal mass.

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

The MPC confinement boundary contains no valves or other pressure relief devices. The MPC enclosure vessel is shown to meet the stress intensity criteria of the ASME Code, Section III, Subsection NB for all service conditions. Therefore, the demonstration that the enclosure vessel meets Subsection NB limits ensures that there is no unacceptable release of radioactive materials.

The HI-STORM 100 storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of unreinforced (plain) concrete. Additional regions of fully confined (by enveloping steel structure) unreinforced concrete are attached to the lid and to the baseplate. The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a cushion for the MPC in the event of a tip-over accident. The storage overpack is not a pressure vessel since it contains cooling vents that do not allow for a differential pressure to develop across the overpack wall. The structural steel components of the HI-STORM 100 Overpack are designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3. A short version of the HI-STORM 100 overpack, designated as the HI-STORM 100S, is introduced in this revision. To accommodate nuclear plants with limited height access, the HI-STORM 100S has a re-configured lid and a lower overall height. There are minor weight redistributions but the overall bounding weight of the system is unchanged. Therefore, structural analyses are revisited if and only if the modified configuration cannot be demonstrated to be bounded by the original calculation. New or modified calculations focused on the HI-STORM 100 are clearly identified within the text of this chapter. Unless otherwise designated, general statements using the terminology "HI-STORM 100" also apply to the HI-STORM 100S. The HI-STORM 100S can carry all MPC's and transfer casks that can be carried in the HI-STORM 100.

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(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.

Table 2.2.6 provides a listing of the applicable design codes for all structures, systems, and components which are designated as ITS.

### 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.

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

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

The above design objectives for the HI-STORM 100 System can be particularized for individual components as follows:

- The objectives of the structural analysis of the MPC are to demonstrate that:
  1. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.
  2. The MPC basket does not deform under credible loading conditions such that the subcriticality or retrievability of the SNF is jeopardized.
- The objectives of the structural analysis of the storage overpack are to demonstrate that:
  1. Tornado-generated missiles do not compromise the integrity of the MPC confinement boundary.
  2. The overpack can safely provide for on-site transfer of the loaded MPC and ensure adequate support to the HI-TRAC transfer cask during loading and unloading of the MPC.
  3. The radiation shielding remains properly positioned in the case of any normal, off-normal, or natural phenomenon or accident event.
  4. The flow path for the cooling air flow shall remain available under normal and off-normal conditions of storage and after a natural phenomenon or accident event.
  - 4.5. The loads arising from normal, off-normal, and accident level conditions exerted on the contained MPC do not exceed the structural design criteria of the MPC.
  6. No geometry changes occur under any normal, off-normal, and accident level conditions of storage that may preclude ready retrievability of the contained MPC.

7. A freestanding storage overpack can safely withstand a non-mechanistic tip-over event with a loaded MPC within the overpack. The HI-STORM 100A is specifically engineered to be permanently attached to the ISFSI pad. The ISFSI pad engineered for the anchored cask is designated as “Important to Safety”. Therefore, the non-mechanistic tipover is not applicable to the HI-STORM 100A.
  8. The inter-cask transfer of a loaded MPC can be carried out without exceeding the structural capacity of the HI-STORM 100 Overpack, provided all required auxiliary equipment and components specific to an ISFSI site comply with their Design Criteria set forth in this FSAR and the handling operations are in full compliance with operational limits and controls prescribed in this FSAR.
- The objective of the structural analysis of the HI-TRAC transfer cask is to demonstrate that:
    1. Tornado generated missiles do not compromise the integrity of the MPC confinement boundary while the MPC is contained within HI-TRAC.
    2. No geometry changes occur under any postulated handling or storage conditions that may preclude ready retrievability of the contained MPC.
    3. The structural components perform their intended function during lifting and handling with the loaded MPC.
    4. The radiation shielding remains properly positioned under all applicable handling service conditions for HI-TRAC.
    5. The lead shielding, top lid, and transfer lid doors remain properly positioned during postulated handling accidents.

The aforementioned objectives are deemed to be satisfied for the MPC, the overpack, and the HI-TRAC, if stresses (or stress intensities, as applicable) calculated by the appropriate structural analyses are less than the allowables defined in Subsection 3.1.2.2, and if the diametral change in the storage overpack (or HI-TRAC), if any, after any event of structural consequence to the overpack (or transfer cask), does not preclude ready retrievability of the contained MPC.

Stresses arise in the components of the HI-STORM 100 System due to various loads that originate under normal, off-normal, or accident conditions. These individual loads are combined to form load combinations. Stresses and stress intensities resulting from the load combinations are compared to their respective allowable stresses and stress intensities. The following subsections present loads, load combinations, and the allowable limits germane to them for use in the structural analyses of the MPC, the overpack, and the HI-TRAC transfer cask.

### 3.1.2.1 Loads and Load Combinations

The individual loads applicable to the HI-STORM 100 System and the HI-TRAC cask are defined in Section 2.2 of this report (Table 2.2.13). Load combinations are developed by assembling the individual loads that may act concurrently, and possibly, synergistically (Table 2.2.14). In this subsection, the individual loads are further clarified as appropriate and the required load combinations are identified. Table 3.1.1 contains the load combinations for the storage overpack where kinematic stability is of primary importance. The load combinations where stress or load level is of primary importance are set forth in Table 3.1.3 for the MPC fuel basket, in Table 3.1.4 for the MPC confinement boundary, and in Table 3.1.5 for the storage overpack and the HI-TRAC transfer cask. Load combinations are applied to the mathematical models of the MPCs, the overpack, and the HI-TRAC. Results of the analyses carried out under bounding load combinations are compared with their respective allowable stresses (or stress intensities, as applicable). The analysis results from the bounding load combinations are also assessed, where warranted, to ensure satisfaction of the functional performance criteria discussed in the preceding subsection.

#### 3.1.2.1.1 Individual Load Cases

The individual loads that address each design criterion applicable to the structural design of the HI-STORM 100 System are catalogued in Table 2.2.13. Each load is given a symbol for subsequent use in the load combination listed in Table 2.2.14.

Accident condition and natural phenomena-induced events, collectively referred to as the "Level D" condition in Section III of the ASME Boiler & Pressure Vessel Codes, in general, do not have a universally prescribed limit. For example, the impact load from a tornado-borne missile, or the overturning load under flood or tsunami, cannot be prescribed as design basis values with absolute certainty that all ISFSI sites will be covered. Therefore, as applicable, allowable magnitudes of such loadings are postulated for the HI-STORM 100 System. The allowable values are drawn from regulatory and industry documents (such as for tornado missiles and wind) or from an intrinsic limitation in the system (such as the permissible "drop height" under a postulated handling accident). In the following, the essential characteristic of each "Level D" type loading is explained.

##### 3.1.2.1.1.1 Tip-Over

It is required to demonstrate that the free-standing HI-STORM 100 storage overpack, containing a loaded MPC, will not tip over as a result of a postulated natural phenomenon event, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tip-over scenario per NUREG-1536 is analyzed. Since the HI-STORM 100S has an overall length that is less than the regular HI-STORM 100, the maximum impact velocity of the overpack will be reduced. Therefore, the results of the tipover analysis for the HI-STORM 100 (reported in Appendix 3.A) are bounding for the HI-STORM 100S. The potential of the HI-STORM 100 Overpack tipping over during the lowering (or raising) of the loaded MPC into (or out of) it with the HI-TRAC cask mounted on it is ruled out because of the safeguards and devices mandated by this FSAR for such operations

(Subsection 2.3.3.1 and Technical Specification 4.9). The physical and procedural barriers under the MPC handling operations have been set down in the FSAR to preclude overturning of the HI-STORM/HI-TRAC assemblage with an extremely high level of certainty. Much of the ancillary equipment needed for the MPC transfer operations must be custom engineered to best accord with the structural and architectural exigencies of the ISFSI site. Therefore, with the exception of the HI-TRAC cask, their design cannot be prescribed, a priori, in this FSAR. However, carefully drafted Design Criteria and conditions of use set forth in this FSAR eliminate the potential of weakening of the safety measures contemplated herein to preclude an overturning event during MPC transfer operations. Subsection 2.3.3.1 contains a comprehensive set of design criteria for the ancillary equipment and components required for MPC transfer operations to ensure that the design objective of precluding a kinematic instability event during MPC transfer operations is met. Further information on the steps taken to preclude system overturning during MPC transfer operations may be found in Chapter 8, Section 8.0.

In the HI-STORM 100A configuration, wherein the overpack is physically anchored to the ISFSI pad, the potential for a tip-over is a priori precluded. Therefore, the ISFSI pad need not be engineered to be sufficiently compliant to limit the peak MPC deceleration to Table 2.2.8 values. The stiffness of the pad, however, may be controlled by the ISFSI structural design and, therefore, may result in a reduced “carry height” from that specified for a free-standing cask. If a non-single failure proof lifting device is employed to carry the cask over the pad, determination of maximum carry height must be performed by the ISFSI owner once the ISFSI pad design is formalized.

#### 3.1.2.1.1.2 Handling Accident

A handling accident during transport of a loaded HI-STORM 100 storage overpack is assumed to result in a vertical drop. The HI-STORM 100 storage overpack will not be handled in a horizontal position while containing a loaded MPC. Therefore, a side drop is not considered a credible event.

HI-TRAC can be carried in a horizontal orientation while housing a loaded MPC. Therefore, a handling accident during transport of a loaded HI-TRAC in a horizontal orientation is considered to be a credible accident event.

As discussed in the foregoing, the vertical drop of the HI-TRAC and the tip-over of the assemblage of a loaded HI-TRAC on the top of the HI-STORM 100 storage overpack during MPC transfer operations do not need to be considered.

#### 3.1.2.1.1.3 Flood

The postulated flood event results into two discrete scenarios which must be considered; namely,

1. stability of the HI-STORM 100 System due to flood water velocity, and
2. structural effects of hydrostatic pressure and water velocity induced lateral pressure.

The maximum hydrostatic pressure on the cask in a flood where the water level is conservatively set at 125 feet is calculated as follows:

Using

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

The maximum hydrostatic pressure is

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

The accident condition design external pressure for the MPC (Table 2.2.1) bounds the maximum hydrostatic pressure exerted by the flood.

#### 3.1.2.1.1.4 Explosion

Explosion, by definition, is a transient event. Explosive materials (except for the short duration when a limited quantity of motive fuel for placing the loaded MPC on the ISFSI pad is present in the tow vehicle) are prohibited in the controlled area by specific stipulation in the HI-STORM 100 Technical Specification. However, pressure waves emanating from explosions in areas outside the ISFSI are credible.

Pressure waves from an explosive blast in a property near the ISFSI site result in an impulsive aerodynamic loading on the stored HI-STORM 100 Overpacks. Depending on the rapidity of the pressure build-up, the inside and outside pressures on the HI-STORM METCON™ shell may not equalize, leading to a net lateral loading on the upright overpack as the pressure wave traverses the overpack. The magnitude of the dynamic pressure wave is conservatively set to a value below the magnitude of the pressure differential that would cause a tip-over of the cask if the pulse duration was set at one second. With the maximum design basis pressure pulse established (by setting the design basis pressure differential sufficiently low that cask tip-over is not credible due to the travelling pressure wave), the stress state under this condition requires analysis. The lateral pressure difference, applied over the overpack full height, causes axial and circumferential stresses and strains to develop. Level D stress limits must not be exceeded under this state of stress. It must also be demonstrated that no permanent ovalization of the cross section occurs that leads to loss of clearance to remove the MPC after the explosion.

Once the pressure wave traverses the cask body, then an elastic stability evaluation is warranted. An all-enveloping pressure from the explosion may threaten safety by buckling the overpack outer shell.

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

The maximum overpressure on the MPC resulting from an explosion is limited by the HI-STORM Technical Specification to be equal to or less than the accident condition design external pressure or external pressure differential specified in Table 2.2.1. The design external pressure differential is applied as a component of the load combinations.

#### 3.1.2.1.1.5 Tornado

The three components of a tornado load are:

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

Wind speeds and tornado-induced pressure drop are specified in Table 2.2.4. Tornado missiles are listed in Table 2.2.5. A central functional objective of a storage overpack is to maintain the integrity of the “confinement boundary”, namely, the multi-purpose canister stored inside it. This operational imperative requires that the mechanical loadings associated with a tornado at the ISFSI do not jeopardize the physical integrity of the loaded MPC. Potential consequences of a tornado on the cask system are:

- Instability (tip-over) due to tornado missile impact plus either steady wind or impulse from the pressure drop (only applicable for free-standing cask).
- Stress in the overpack induced by the lateral force caused by the steady wind or missile impact.
- Loadings applied on the MPC transmitted to the inside of the overpack through its openings or as a secondary effect of loading on the enveloping overpack structure.
- Excessive storage overpack permanent deformation that may prevent ready retrievability of the MPC.
- Excessive storage overpack permanent deformation that may significantly reduce the shielding effectiveness of the storage overpack.

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

- The loaded overpack does not become kinematically unstable (only applicable for free-standing cask).



- The overpack does not deform plastically such that the retrievability of the stored MPC is threatened.
- The MPC does not sustain an impact from an incident missile.
- The MPC is not subjected to inertia loads (acceleration or deceleration) in excess of its design basis limit set forth in Chapter 2 herein.
- The overpack does not deform sufficiently due to tornado-borne missiles such that the shielding effectiveness of the overpack is significantly affected.

The results obtained for the HI-STORM 100 bound the corresponding results for HI-STORM 100S because of the reduced height. In the anchored configuration (HI-STORM 100A), the kinematic stability requirement stated above is replaced with the requirement that the stresses in the anchor studs do not exceed level D stress limits for ASME Section III, Class 3, Subsection NF components.

#### 3.1.2.1.1.6 Earthquake

Subsections 2.2.3.7 and 3.4.7 contain the detailed specification of the seismic inputs applied to the HI-STORM 100 System. The design basis earthquake is assumed to be at the top of the ISFSI pad. Potential consequences of a seismic event are sliding/overturning of a free-standing cask, overstress of the sector lugs and anchor studs for the anchored HI-STORM 100A, and lateral force on the overpack causing excessive stress and deformation of the storage overpack.

In the anchored configuration (HI-STORM 100A), a seismic event results in a fluctuation in the state of stress in the anchor bolts and a local bending action on the sector lugs.

Analyses must be performed to ensure that:

- The maximum axial stress in the anchor bolts remains below the Level D stress limits for Section III Class 3 Subsection NF components.
- The maximum primary membrane plus bending stress intensity in the sector lugs during the DBE event satisfies Level D stress limits of the ASME Code, Subsection NF.
- The anchor bolts will not sustain fatigue failure due to pulsation in their axial stress during the DBE event.
- The stress in the weld line joining the sector lugs to the HI-STORM 100 weldment is within Subsection NF limits for Level D condition.

#### 3.1.2.1.1.7 Lightning

The HI-STORM 100 Overpack contains over 25,000 lb of highly conductive carbon steel with over 700 square feet of external surface area. Such a large surface area and metal mass is adequate to dissipate any lightning that may strike the HI-STORM 100 System. There are no combustible materials on the HI-STORM 100 surface. Therefore, lightning will not impair the structural performance of components of the HI-STORM 100 System that are important to safety.

#### 3.1.2.1.1.8 Fire

The potential structural consequences of a fire are: the possibility of an interference developing between the storage overpack and the loaded MPC due to free thermal expansion; and, the degradation of material properties to the extent that their structural performance is affected during a subsequent recovery action. The fire condition is addressed to the extent necessary to demonstrate that these adverse structural consequences do not materialize.

#### 3.1.2.1.1.9 100% Fuel Rod Rupture

The effect on structural performance by 100% fuel rod rupture is felt as an increase in internal pressure. The accident internal pressure limit set in Chapter 2 bounds the pressure from 100% fuel rod rupture. Therefore, no new load condition has been identified.

#### 3.1.2.1.2 Load Combinations

Load combinations are created by summing the effects of several individual loads. The load combinations are selected for the normal, off-normal, and accident conditions. The loadings appropriate for HI-STORM 100 under the various conditions are presented in Table 2.2.14. These loadings are combined into meaningful combinations for the various HI-STORM 100 System components in Tables 3.1.1, and 3.1.3-3.1.5. Table 3.1.1 lists the load combinations that address overpack stability. Tables 3.1.3 through 3.1.5 list the applicable load combinations for the fuel basket, the enclosure vessel, and the overpack and HI-TRAC, respectively.

As discussed in Subsection 2.2.7, the number of discrete load combinations for each situational condition (i.e., normal, off-normal, etc.) is consolidated by defining bounding loads for certain groups of loadings. Thus, the accident condition pressure  $P_o^*$  bounds the surface loadings arising from accident and extreme natural phenomenon events, namely, tornado wind  $W'$ , flood  $F$ , and explosion  $E^*$ .

As noted previously, certain loads, namely earthquake  $E$ , flowing water under flood condition  $F$ , force from an explosion pressure pulse  $F^*$ , and tornado missile  $M$ , act to destabilize a cask. Additionally, these loads act on the overpack and produce essentially localized stresses at the HI-STORM 100 System to ISFSI interface. Table 3.1.1 provides the load combinations that are relevant to the stability analyses of free-standing casks. The site ISFSI DBE zero period acceleration (ZPA) must be bounded by the design basis seismic ZPA defined by the Load Combination C of Table 3.1.1 to demonstrate that the margin against tip-over during a seismic event is maintained.

The major constituents in the HI-STORM 100 System are: (i) the fuel basket, (ii) the enclosure vessel, (iii) the HI-STORM 100 (or HI-STORM 100S) Overpack, and (iv) the HI-TRAC transfer cask. The fuel basket and the enclosure vessel (EV) together constitute the multi-purpose canister. The multi-purpose canister (MPC) is common to HI-STORM 100 and HI-STAR 100, and as such, has been extensively analyzed in the storage FSAR and transport SAR (Dockets 72-1008 and 71-9261) for HI-STAR 100. Many of the loadings on the MPC (fuel basket and enclosure vessel) are equal to or bounded by loadings already considered in the HI-STAR 100 SAR documents. Where such analyses have been performed, their location in the HI-STAR 100 SAR documents is indicated in this HI-STORM 100 SAR for continuity in narration. A complete account of analyses and results for all load combinations for all four constituents parts is provided in Section 3.4 as required by Regulatory Guide 3.61.

In the following, the loadings listed as applicable for each situational condition in Table 2.2.14 are addressed in meaningful load combinations for the fuel basket, enclosure vessel, and the overpack. Each component is considered separately.

### Fuel Basket

Table 3.1.3 summarizes all loading cases (derived from Table 2.2.14) that are germane to demonstrating compliance of the fuel baskets to Subsection NG when these baskets are housed within HI-STORM 100 or HI-TRAC.

The fuel basket is not a pressure vessel; therefore, the pressure loadings are not meaningful loads for the basket. Further, the basket is structurally decoupled from the enclosure vessel. The gap between the basket and the enclosure vessel is sized to ensure that no constraint of free-end thermal expansion of the basket occurs. The demonstration of the adequacy of the basket-to-the-enclosure vessel (EV) gap to ensure absence of interference is a physical problem that must be analyzed.

The normal handling loads on the fuel basket in an MPC within the HI-STORM 100 System or the HI-TRAC transfer cask are identical to or bounded by the normal handling loads analyzed in the HI-STAR 100 FSAR Docket Number 72-1008.

Three accident condition scenarios must be considered: (i) drop with the storage overpack axis vertical; (ii) drop with the HI-TRAC axis horizontal; and (iii) storage overpack tipover. The vertical drop scenario is considered in the HI-STAR 100 SAR.

The horizontal drop and tip-over must consider multiple orientation of the fuel basket, as the fuel basket is not radially symmetric. Therefore, two horizontal drop orientations are considered which are referred to as the 0 degree drop and 45 degree drop, respectively. In the 0 degree drop, the basket drops with its panels oriented parallel and normal to the vertical (see Figure 3.1.2). The 45-degree drop implies that the basket's honeycomb section is rotated meridionally by 45 degrees (Figure 3.1.3).

## Enclosure Vessel

Table 3.1.4 summarizes all load cases that are applicable to structural analysis of the enclosure vessel to ensure integrity of the confinement boundary.

The enclosure vessel is a pressure vessel consisting of a cylindrical shell, a thick circular baseplate at the bottom, and a thick circular lid at the top. This pressure vessel must be shown to meet the primary stress intensity limits for ASME Section III Class 1 at the design temperature and primary plus secondary stress intensity limits under the combined action of pressure plus thermal loads.

Normal handling of the enclosure vessel is considered in Docket 72-1008; the handling loads are independent of whether the enclosure vessel is within HI-STAR 100, HI-STORM 100, or HI-TRAC.

The off-normal condition handling loads are identical to the normal condition and, therefore, a separate analysis is not required.

Analyses presented in this chapter are intended to demonstrate that the maximum decelerations in drop and tip-over accident events are limited by the bounding values in Table 3.1.2. The vertical drop event is considered in the HI-STAR 100 SAR Docket 72-1008.

The deceleration loadings developed in the enclosure vessel during a horizontal drop event are combined with those due to  $P_i$  (internal pressure) acting alone. The accident condition pressure is bounded by  $P_i^*$ . The design basis deceleration for the MPC in the HI-STAR 100 System is 60g's, whereas the design basis deceleration for the MPC in the HI-STORM 100 System is 45g's. The design pressures are identical. The fire event ( $T^*$  loading) is considered for ensuring absence of interference between the enclosure vessel and the fuel basket and between the enclosure vessel and the overpack.

It is noted that the MPC basket-enclosure vessel thermal expansion and stress analyses are reconsidered in this submittal to reflect the different MPC-to-overpack gaps that exist in the HI-STORM 100 Overpack versus the HI-STAR 100 overpack, coupled with the different design basis decelerations.

## Storage Overpack

Table 3.1.5 identifies the load cases to be considered for the overpack. These are in addition to the kinematic criteria listed in Table 3.1.1. Within these load cases and kinematic criteria, the following items must be addressed:

### Normal Conditions

- The dead load of the HI-TRAC with the heaviest loaded MPC (dry) on top of the HI-STORM 100 Overpack must be shown to be able to be supported by the metal-concrete (METCON™) structure consisting of the two concentric steel shells and the steel rib plates, and by the concrete columns away from the vent regions.
- The dead load of the HI-STORM 100 Overpack itself must be supportable by the steel structure with no credit for concrete strength other than self-support in compression.
- Normal handling loads must be accommodated without taking any strength credit from the contained concrete other than self-support in compression.

### Accident Conditions

- Maximum flood water velocity for the overpack with an empty MPC must be specified to ensure that no sliding or tip-over occurs.
- Tornado missile plus wind on an overpack with an empty MPC must be specified to demonstrate that no cask tip-over occurs.
- Tornado missile penetration analysis must demonstrate that the postulated large and penetrant missiles cannot contact the MPC. The small missile must be shown not to penetrate the MPC pressure vessel boundary, since it can enter the overpack cavity through the vent ducts.
- Under seismic conditions, a fully loaded, free-standing HI-STORM 100 overpack must be demonstrated to not tip over under the maximum ZPA event. The maximum sliding of the overpack must demonstrate that casks will not impact each other.
- Under a non-mechanistic postulated tip-over of a fully loaded, free-standing HI-STORM 100 overpack, the overpack lid must not dislodge.
- Accident condition stress levels must not be exceeded in the steel and compressive stress levels in the concrete must remain within allowable limits.
- Accident condition induced gross general deformations of the storage overpack must be limited to values that do not preclude ready retrievability of the MPC.

As noted earlier, analyses performed using the HI-STORM 100 generally provide results that are identical to or bound results for the shorter HI-STORM 100S; therefore, analyses are not repeated specifically for the HI-STORM 100S unless the specific geometry changes significantly influence the safety factors.

## HI-TRAC Transfer Cask

Table 3.1.5 identifies load cases applicable to the HI-TRAC transfer cask.

The HI-TRAC transfer cask must provide radiation protection, must act as a handling cask when carrying a loaded MPC, and in the event of a postulated accident must not suffer permanent deformation to the extent that ready retrievability of the MPC is compromised. This submittal includes three types of transfer casks: a 125-ton HI-TRAC (referred to as the HI-TRAC 125), a modified version of the HI-TRAC 125 called the HI-TRAC 125D, and a 100-ton HI-TRAC. The details of these three transfer casks are provided in the design drawings in Section 1.5. The same steel structures (i.e., shell thicknesses, lid thicknesses, etc.) are maintained with the only major differences being in the amount of lead shielding, the water jacket configuration, the bottom flange, and the lower dead weight loading. Therefore, all structural analyses performed for the HI-TRAC 125 are repeated for the HI-TRAC 125D and the HI-TRAC 100 only if it cannot be clearly demonstrated that the HI-TRAC 125 calculation is bounding.

### 3.1.2.2 Allowables

The important to safety components of the HI-STORM 100 System are listed in Table 2.2.6. Allowable stresses, as appropriate, are tabulated for these components for all service conditions.

In Subsection 2.2.5, the applicable service level from the ASME Code for determination of allowables is listed. Table 2.2.14 provides a tabulation of normal, off-normal, and accident conditions and the service levels defined in the ASME Code, along with the applicable loadings for each service condition.

Allowable stresses and stress intensities are calculated using the data provided in the ASME Code and Tables 2.2.10 through 2.2.12. Tables 3.1.6 through 3.1.16 contain numerical values of the stresses/stress intensities for all MPC, overpack, and HI-TRAC load bearing materials as a function of temperature.

In all tables the terms  $S$ ,  $S_m$ ,  $S_y$ , and  $S_u$ , respectively, denote the design stress, design stress intensity, minimum yield strength, and the ultimate strength. Property values at intermediate temperatures that are not reported in the ASME Code are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the Code in any structural calculation.

Additional terms relevant to the analyses are extracted from the ASME Code (Figure NB-3222-1, for example) as follows:

Symbol	Description	Notes
$P_m$	Average primary stress across a solid section	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.
$P_L$	Average stress across any solid section	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
$P_b$	Primary bending stress	Component of primary stress proportional to the distance from the centroid of a solid section. Excludes the effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects.
$P_e$	Secondary expansion stress	Stresses that result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration. (Not applicable to vessels.)
$Q$	Secondary membrane plus bending stress	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion.
$F$	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses 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 that 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 lift devices in the HI-STORM 100 Overpack and HI-TRAC casks and the multi-purpose canisters, collectively referred to as "trunnions", are subject to specific limits set forth by NUREG-0612: the primary stresses in a trunnion must be less than the smaller of 1/10 of the material ultimate strength and 1/6 of the material yield strength under a normal handling condition (Load 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 normal storage mode, the lowest service temperature (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 parts used to lift the overpack or transfer cask (see Table 2.2.2 and Chapter 12). This includes the anchor block in the HI-STORM 100 Overpack, and pocket trunnions, lifting trunnions and the lifting trunnion block in HI-TRAC. Such items will henceforth be referred to as "significant-to-handling" (STH) parts. The applicable code for these elements of the structure is ANSI N14.6.

LST = -40°F for all HI-STORM "NF" components and 0°F for all HI-TRAC "NF" components. This includes all "NF" items not identified as an STH part.



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 toughness, but little effect on tensile or yield strength of the material, are also not specified with the degree of specificity in the ASME Code to guarantee a well defined fracture toughness. In fact, the Code relies on actual coupon testing of the part to ensure the desired level of protection against brittle fracture. For Section III, Subsection NF Class 3 parts, the desired level of protection is considered to exist if the lowest service temperature is equal to or greater than the NDT temperature (per NF 2311(b)(10)). Accordingly, the required NDT temperature for all load bearing metal parts in the HI-STORM 100 Overpack (“NF” and “STH”) is -40°F. Likewise, the NDT temperature for all “NF” parts in HI-TRAC (except for “STH” parts) is set equal to 0°F.

From the standpoint of protection against brittle fracture, it should be recognized that setting the LST equal to the NDT temperature ensures that the fracture strength of the material containing small flaws is equal to its yield strength. In fact, as the stress calculations in this chapter ~~(and associated appendices)~~ would attest, the maximum primary tensile stress in the HI-STORM 100 Overpack is below 6,000 psi in all normal conditions of storage operating modes. Even in extreme environmental phenomena events, tensile stresses are below 6,000 psi, except for localized regions under postulated missile impacts or non-mechanistic tip-over. For ferritic steels (please see NF-2311(b)(7)), 6,000 psi is the threshold stress, at or below which crack propagation will not take place, no matter how low the metal temperature [3.1.3, p. 13]. (The threshold stress is the horizontal extension of the crack arrest temperature (CAT) curve in the fracture mechanics literature.)

The generally low value of tensile stress in the HI-STORM 100 storage overpack and in the HI-TRAC cask parts suggest that an NDT temperature requirement is not essential to ensure safety from crack growth. However, the aforementioned NDT temperature requirement of -40°F has been imposed to incorporate an additional layer of conservatism in the design.

The STH components (bolt anchor block (HI-STORM), lifting trunnion (HI-TRAC), lifting trunnion block (HI-TRAC), and pocket trunnion (HI-TRAC) have thicknesses greater than 2". SA350-LF3 has been selected as the material for these items (except for the lifting trunnions) due to its capability to maintain acceptable fracture toughness at low temperatures (see Table 5 in SA350 of ASME Section

IIA). Additionally, material for the HI-TRAC top flange, pool lid (100 ton) and pool lid outer ring (125 ton) has been defined as SA350-LF3, SA350-LF2, or SA203E (see Table A1.15 of ASME Section IIA) in order to achieve low temperature fracture toughness. The HI-TRAC lifting trunnion is fabricated from SB-637 Grade N07718, a high strength nickel alloy material. This material has a high resistance to fracture at low temperatures. All other steel structural materials in the HI-STORM 100 overpack and HI-TRAC cask are made of SA516-70 or SA515-70 (with some components having an option for SA203E or SA350-LF3 depending on material availability).

Table 3.1.18 provides a summary of impact testing requirements to satisfy the requirements for prevention of brittle fracture.

#### 3.1.2.4 Fatigue

In storage, the HI-STORM 100 System is not subject to significant cyclic loads. Failure due to fatigue is not a concern for the HI-STORM 100 System.

In an anchored installation, however, the anchor studs sustain a pulsation in the axial load during the seismic event. The amplitude of axial stress variation under the DBE event is computed in this chapter and a significant margin of safety against fatigue failure during the DBE event is demonstrated.

The system is subject to cyclic temperature fluctuations. These fluctuations result in small changes of thermal expansions and pressures in the MPC. The loads resulting from these changes are small and do not significantly contribute to the "usage factor" of the cask.

Inspection of the HI-TRAC trunnions specified in Chapter 9 will preclude use of a trunnion that exhibits visual damage.

#### 3.1.2.5 Buckling

Certain load combinations subject structural sections with relatively large slenderness ratios (such as the enclosure vessel shell) to compressive stresses that may actuate buckling instability before the allowable stress is reached. Tables 3.1.4 and 3.1.5 list load combinations for the enclosure vessel and the HI-STORM 100/HI-TRAC structures; the cases which warrant stability (buckling) check are listed therein (note that a potential buckling load has already been identified as a consequence of a postulated explosion).

**TABLE 3.1.1**

**LOAD COMBINATIONS SIGNIFICANT TO HI-STORM 100 OVERPACK  
KINEMATIC STABILITY ANALYSIS**

<b>Loading Case</b>	<b>Combinations<sup>†</sup></b>	<b>Comment</b>	<b>Analysis of this Load Case Presented in:</b>
A	D + F	This case establishes flood water flow velocity with a minimum safety factor of 1.1 against overturning and sliding.	Subsection 3.4.6
B	D + M + W'	Demonstrate that the HI-STORM 100 Overpack with minimum SNF stored (minimum D) will not tip over.	<i>Subsection 3.4.8</i> <del>Appendix 3.C</del>
C	D + E	Establish the value of ZPA <sup>††</sup> that will not cause the overpack to tip over.	Subsection 3.4.7

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<sup>†</sup> Loading symbols are defined in Table 2.2.13

<sup>††</sup> ZPA is zero period acceleration

**TABLE 3.1.2**

**DESIGN BASIS DECELERATIONS FOR THE DROP EVENTS**

<b>Case</b>	<b>Value<sup>†</sup> (in multiples of acceleration due to gravity)</b>
Vertical axis drop (HI-STORM 100 Overpack only)	45
Horizontal axis (side) drop (HI-TRAC only)	45

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<sup>†</sup> The design basis value is set from the requirements of the HI-STORM 100 System, as its components are operated as a storage system. The MPC is designed to higher loadings (60g's vertical and horizontal) when in a HI-STAR 100 overpack. Analysis of the MPC in a HI-STAR 100 overpack under a 60g loading is provided in HI-STAR 100 Docket Numbers 71-9261 and 72-1008.

**TABLE 3.1.3**

**LOADING CASES FOR THE FUEL BASKET**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location Where this Case is Evaluated</b>
F1	T, T'	Demonstrate that the most adverse of the temperature distributions in the basket will not cause fuel basket to expand and contact the enclosure vessel wall. Compute the secondary stress intensity and show that it is small.	Appendices 3-I, 3-J, 3-U, 3-V, 3-W; Subsection 3.4.4.2
F2 (Note 1)	D + H	Conservatively add the stresses in the basket due to vertical and horizontal orientation handling to form a bounding stress intensity.	Section 3.4 <del>Section</del> Docket 72-1008
F3 F3.a (Note 2)	D + H'	Vertical axis drop event	Docket Number 72-1008, Subsection 3.4.4.3.1.6
F3.b (Note 3)	D + H'	Side Drop, 0 degree orientation (Figure 3.1.2)	Table 3.4.6
F3.c (Note 3)	D + H'	Side Drop, 45 degree orientation (Figure 3.1.3)	Table 3.4.6

Notes:

1. Load Case F2 for the HI-STORM 100 System is identical to Load Case F2 for the HI-STAR 100 System in Docket Number 72-1008, Table 3.1.3.
2. Load Case F3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008, Subsection 3.4.4.3.1.6. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases F3.b and F3.c are analyzed here for a 45g deceleration, while the MPC is housed within a HI-STORM 100 Overpack or a HI-TRAC transfer cask. The initial clearance at the interface between the MPC shell and the HI-STORM 100 Overpack or HI-TRAC transfer cask is greater than or equal to the initial clearance between the MPC shell and the HI-STAR 100 overpack. This difference in clearance directly affects the stress field. The side drop analysis for the MPC in the HI-STAR 100 overpack under 60g's bounds the corresponding analysis of the MPC in HI-TRAC for 45 g's.

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<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.1.4**

**LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†</sup></b>	<b>Notes</b>	<b>Comments and Location Where this Case is Analyzed</b>
E1 (Note 1)			
E1.a	Design internal pressure, P <sub>i</sub>	Primary stress intensity limits in the shell, baseplate, and closure ring	E1.a Lid Docket 72-1008 3.E.8.1.1 Baseplate Docket 72-1008 3.I.8.1 Shell 3.4.4.3.1.2 Supports N/A
E1.b	Design external pressure, P <sub>o</sub>	Primary stress intensity limits, buckling stability	E1.b Lid P <sub>i</sub> bounds Baseplate P <sub>i</sub> bounds Shell Docket 72-1008 – <i>Buckling Methodology in 3.H-3.H (Case 4)</i> Supports N/A
E1.c	Design internal pressure, P <sub>i</sub> , Plus Temperature, T	Primary plus secondary stress intensity under Level A condition	E1.c Lid, Baseplate, and Shell Section 3.4.4.3.1.2
E2	D + H + (P <sub>i</sub> , P <sub>o</sub> ) <sup>††</sup>	Vertical lift, internal operating pressure conservatively assumed to be equal to the normal design pressure. Principal area of concern is the lid assembly.	Lid Docket 72-1008 3.E.8.1.2 Baseplate Docket 72-1008 3.I.8.2 Shell Docket 72-1008 <del>Section 3.43.AA (stress)</del> Docket 72-1008 <i>Buckling (methodology in 3.H of Docket 72-1008)</i> <del>3.H (Case 4) (buckling)</del> Supports Docket 72-1008 <del>Section 3.43.AA</del>

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

<sup>††</sup> The notation (P<sub>i</sub>, P<sub>o</sub>) means that both cases are checked with either P<sub>o</sub> or P<sub>i</sub> applied.

**TABLE 3.1.4 (CONTINUED)**

**LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

Load Case I.D.	Load Combination <sup>†</sup>	Notes	Comments and Location Where this Case is Analyzed
E3 E3.a (Note 2)  E3.b (Note 3)  E3.c (Note 3)	$D + H' + (P_o, P_i)$  $D + H' + (P_i, P_o)$  $D + H' + (P_i, P_o)$	Vertical axis drop event  Side drop, 0 degree orientation (Figure 3.1.2)  Side drop, 45 degree orientation (Figure 3.1.3)	E3.a Lid Docket 72-1008 3.E.8.2.1-2 Baseplate Docket 72-1008 3.I.8.3 Shell <i>Buckling (methodology in 3.H of Docket 72-1008)</i> <del>Docket 72-1008 3.H (Case 5) (Buckling)</del> Supports N/A  E3.b Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6, <del>3.Y</del>  E3.c Lid End drop bounds Baseplate End drop bounds Shell Table 3.4.6 Supports Table 3.4.6, <del>3.Y</del>
E4	T	Demonstrate that interference with the overpack will not develop for T.	Section 3.4.4.2

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.1.4 (CONTINUED)**

**LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)**

Load Case I.D.	Load Combination <sup>†</sup>	Notes	Comments and Location Where this Case is Analyzed	
E5 (Note 1)	$P_i^*$ or $P_o^*$ + D + T*	Demonstrate compliance with level D stress limits – buckling stability.	Lid Baseplate Shell <del>Docket 72-1008</del> Supports	Docket 72-1008 3.E.8.2.1.3 Docket 72-1008 3.I.8.4 <b><i>Buckling (methodology in 3.H of Docket 72-1008)</i></b> <del>3.H (Case 6) (buckling)</del> Docket 72-1008 3.4.4.3.1.5 (thermal stress) N/A

Notes:

1. Load Cases E1.a, E1.b, E2, and E5 are identical to the load cases presented in Docket Number 72-1008, Table 3.1.4. Design pressures and MPC weights are identical.
2. Load Case E3.a is bounded by the 60g deceleration analysis performed for the HI-STAR 100 System in Docket Number 72-1008, *Section 3.4 Appendix 3-AA*. The HI-STORM 100 vertical deceleration loading is limited to 45g.
3. Load Cases E3.b and E3.c are analyzed in this HI-STORM 100 SAR for a 45g deceleration, while the MPC is housed within the HI-STORM 100 storage overpack. The interface between the MPC shell and storage overpack is not identical to the MPC shell and HI-STAR 100 overpack. The analysis for an MPC housed in HI-TRAC is not performed since results are bounded by those reported in the HI-STAR 100 TSAR for a 60g deceleration.

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.



**TABLE 3.1.5**

**LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location in FSAR—Where this Case is Analyzed</b>
01	D + H + T + (P <sub>o</sub> ,P <sub>i</sub> )	Vertical load handling of HI-STORM 100 Overpack/HI-TRAC.	Overpack <del>3.4.3.53-D</del>  HI-TRAC Shell 3.4.3.3, 3.4.3.43-AE Pool lid 3.4.3.83-AB Transfer lid 3.4.3.93-AD
02			
02.a	D + H' + (P <sub>o</sub> ,P <sub>i</sub> )	Storage Overpack: End drop; primary stress intensities must meet level D stress limits.	Overpack 3.4.4.3.2.3 <del>3-M</del>
02.b	D + H' + (P <sub>o</sub> ,P <sub>i</sub> )	HI-TRAC: Horizontal (side) drop; meet level D stress limits for NF Class 3 components away from the impacted zone; show lids stay in-place. Show primary and secondary impact decelerations are within design basis. (This case is only applicable to HI-TRAC.)	HI-TRAC Shell 3.4.9.13-Z Transfer Lid 3.4.4.3.3.3 <del>3-AD</del> Slapdown 3.4.9.23-AN
02.c	D + H'	Storage Overpack: Tip-over; any permanent deformations must not preclude ready retrieval of the MPC.	Overpack 3.4.10, 3.A <del>3-B</del>

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13

**TABLE 3.1.5 (CONTINUED)**

**LOAD CASES FOR THE HI-STORM 100 OVERPACK/HI-TRAC TRANSFER CASK**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Notes</b>	<b>Location in FSAR—Where this Case is Analyzed</b>
03	D (water jacket)	Satisfy primary membrane plus bending stress limits for water jacket (This case is only applicable to HI-TRAC).	<del>3.4.4.3.3.4</del> <del>3.AG</del>
04	M (penetrant missiles)	Demonstrate that no thru-wall breach of the HI-STORM overpack or HI-TRAC transfer cask occurs, and the primary stress levels are not exceeded. Small and intermediate missiles are examined for HI-STORM and HI-TRAC. Large missile penetration is also examined for HI-TRAC.	Overpack <del>3.4.8.13.G</del> HI-TRAC <del>3.4.8.2.1,</del> <del>3.4.8.2.2</del> <del>3.AN, 3.H</del>
05	P <sub>o</sub>	Explosion: must not produce buckling or exceed primary stress levels in the overpack structure.	<del>3.4.4.5.2,</del> <del>3.4.7.23.B,</del> <del>3.AK</del>

Notes:

1. Under each of these load cases, different regions of the structure are analyzed to demonstrate compliance.

---

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13

**TABLE 3.1.6**

**DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA203-E  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	23.3	23.3	35.0	35.0	69.9	69.9
300	23.3	23.3	35.0	35.0	69.9	69.9
400	22.9	22.9	34.4	34.4	68.7	68.7
500	21.6	21.6	32.4	32.4	64.8	64.8

Definitions:

- $S_m$  = Stress intensity values per ASME Code
- $P_m$  = Primary membrane stress intensity
- $P_L$  = Local membrane stress intensity
- $P_b$  = Primary bending stress intensity
- $P_e$  = Expansion stress
- $Q$  = Secondary stress
- $P_L + P_b$  = Either primary or local membrane plus primary bending

Definitions for Table 3.1.6 apply to all following tables unless modified.

Notes:

1. Limits on values are presented in Table 2.2.10.

---

<sup>†</sup> Evaluation required for Design condition only.

<sup>††</sup> Evaluation required for Levels A and B only.  $P_e$  not applicable to vessels.

**TABLE 3.1.7**

**LEVEL D: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA203-E  
**Service Condition:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>		
	<b>P<sub>m</sub></b>	<b>P<sub>L</sub></b>	<b>P<sub>L</sub> + P<sub>b</sub></b>
-20 to 100	49.0	70.0	70.0
200	49.0	70.0	70.0
300	49.0	70.0	70.0
400	48.2	68.8	68.8
500	45.4	64.9	64.9

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed  $0.42 S_u$ .
3. Limits on values are presented in Table 2.2.10.
4.  $P_m$ ,  $P_L$ , and  $P_b$  are defined in Table 3.1.6.

**TABLE 3.1.8****DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA350-LF3  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

<b>Temp. (Deg.F)</b>	<b>Classification and Value (ksi)</b>					
	<b>S<sub>m</sub></b>	<b>P<sub>m</sub><sup>†</sup></b>	<b>P<sub>L</sub><sup>†</sup></b>	<b>P<sub>L</sub> + P<sub>b</sub><sup>†</sup></b>	<b>P<sub>L</sub> + P<sub>b</sub> + Q<sup>††</sup></b>	<b>P<sub>e</sub><sup>††</sup></b>
-20 to 100	23.3	23.3	35.0	35.0	69.9	69.9
200	22.8	22.8	34.2	34.2	68.4	68.4
300	22.2	22.2	33.3	33.3	66.6	66.6
400	21.5	21.5	32.3	32.3	64.5	64.5
500	20.2	20.2	30.3	30.3	60.6	60.6
600	18.5	18.5	27.75	27.75	55.5	55.5
700	16.8	16.8	25.2	25.2	50.4	50.4

## Notes:

1. Source for S<sub>m</sub> is ASME Code
2. Limits on values are presented in Table 2.2.10.
3. S<sub>m</sub>, P<sub>m</sub>, P<sub>L</sub>, P<sub>b</sub>, Q, and P<sub>e</sub> are defined in Table 3.1.6.

---

<sup>†</sup> Evaluation required for Design condition only.

<sup>††</sup> Evaluation required for Levels A and B conditions only. P<sub>e</sub> not applicable to vessels.

**TABLE 3.1.9**

**LEVEL D, STRESS INTENSITY**

**Code:** ASME NB  
**Material:** SA350-LF3  
**Service Conditions:** Level D  
**Item:** Stress Intensity

Temp. (Deg.F)	Classification and Value (ksi)		
	$P_m$	$P_L$	$P_L + P_b$
-20 to 100	49.0	70.0	70.0
200	48.0	68.5	68.5
300	46.7	66.7	66.7
400	45.2	64.6	64.6
500	42.5	60.7	60.7
600	38.9	58.4	58.4
700	35.3	53.1	53.1

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed  $0.42 S_u$ .
3. Limits on values are presented in Table 2.2.10.
4.  $P_m$ ,  $P_L$ , and  $P_b$  are defined in Table 3.1.6.

**TABLE 3.1.10**

**DESIGN AND LEVEL A: STRESS**

**Code:** ASME NF  
**Material:** SA516, Grade 70, SA350-LF3, SA203-E  
**Service Conditions:** Design and Level A  
**Item:** Stress

<b>Temp. (Deg.F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S</b>	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	17.5	17.5	26.3
700	16.6	16.6	24.9

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Limits on values are presented in Table 2.2.12.

**TABLE 3.1.11**

**LEVEL B: STRESS**

**Code:** ASME NF  
**Material:** SA516, Grade 70, SA350-LF3, and SA203-E  
**Service Conditions:** Level B  
**Item:** Stress

<b>Temp. (Deg.F)</b>	<b>Classification and Value (ksi)</b>	
	<b>Membrane Stress</b>	<b>Membrane plus Bending Stress</b>
-20 to 650	23.3	34.9
700	22.1	33.1

Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.10.



**TABLE 3.1.12**

**LEVEL D: STRESS INTENSITY**

**Code:** ASME NF  
**Material:** SA516, Grade 70  
**Service Conditions:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg.F)</b>	<b>Classification and Value (ksi)</b>		
	<b>S<sub>m</sub></b>	<b>P<sub>m</sub></b>	<b>P<sub>m</sub> + P<sub>b</sub></b>
-20 to 100	23.3	45.6	68.4
200	23.1	41.5	62.3
300	22.5	40.4	60.6
400	21.7	39.1	58.7
500	20.5	36.8	55.3
600	18.7	33.7	50.6
650	18.4	33.1	49.7
700	18.3	32.9	49.3

Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.
2. S<sub>m</sub> = Stress intensity values per Table 2A of ASME, Section II, Part D.
3. Limits on values are presented in Table 2.2.12.
4. P<sub>m</sub> and P<sub>b</sub> are defined in Table 3.1.6.

**TABLE 3.1.13**

**DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** Alloy X  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

Temp. (Deg.F)	Classification and Numerical Value					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	20.0	20.0	30.0	30.0	60.0	60.0
200	20.0	20.0	30.0	30.0	60.0	60.0
300	20.0	20.0	30.0	30.0	60.0	60.0
400	18.7	18.7	28.1	28.1	56.1	56.1
500	17.5	17.5	26.3	26.3	52.5	52.5
600	16.4	16.4	24.6	24.6	49.2	49.2
650	16.0	16.0	24.0	24.0	48.0	48.0
700	15.6	15.6	23.4	23.4	46.8	46.8
750	15.2	15.2	22.8	22.8	45.6	45.6
800	14.9	14.9	22.4	22.4	44.7	44.7

Notes:

1.  $S_m$  = Stress intensity values per Table 2A of ASME II, Part D.
2. Alloy X  $S_m$  values are the lowest values for each of the candidate materials at temperature.
3. Stress classification per NB-3220.
4. Limits on values are presented in Table 2.2.10.
5.  $P_m$ ,  $P_L$ ,  $P_b$ ,  $Q$ , and  $P_e$  are defined in Table 3.1.6.

<sup>†</sup> Evaluation required for Design condition only.

<sup>††</sup> Evaluation required for Levels A, B conditions only.  $P_e$  not applicable to vessels.

**TABLE 3.1.14**

**LEVEL D: STRESS INTENSITY**

**Code:** ASME NB  
**Material:** Alloy X  
**Service Conditions:** Level D  
**Item:** Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	P <sub>m</sub>	P <sub>L</sub>	P <sub>L</sub> + P <sub>b</sub>
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

Notes:

1. Level D stress intensities per ASME NB-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S<sub>u</sub>.
3. Limits on values are presented in Table 2.2.10.
4. P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.6.

**TABLE 3.1.15****DESIGN, LEVELS A AND B: STRESS INTENSITY**

**Code:** ASME NG  
**Material:** Alloy X  
**Service Conditions:** Design, Levels A and B  
**Item:** Stress Intensity

<b>Temp. (Deg. F)</b>	<b>Classification and Value (ksi)</b>				
	<b>S<sub>m</sub></b>	<b>P<sub>m</sub></b>	<b>P<sub>m</sub>+P<sub>b</sub></b>	<b>P<sub>m</sub>+P<sub>b</sub> +Q</b>	<b>P<sub>e</sub></b>
-20 to 100	20.0	20.0	30.0	60.0	60.0
200	20.0	20.0	30.0	60.0	60.0
300	20.0	20.0	30.0	60.0	60.0
400	18.7	18.7	28.1	56.1	56.1
500	17.5	17.5	26.3	52.5	52.5
600	16.4	16.4	24.6	49.2	49.2
650	16.0	16.0	24.0	48.0	48.0
700	15.6	15.6	23.4	46.8	46.8
750	15.2	15.2	22.8	45.6	45.6
800	14.9	14.9	22.4	44.7	44.7

## Notes:

1. S<sub>m</sub> = Stress intensity values per Table 2A of ASME, Section II, Part D.
2. Alloy X S<sub>m</sub> values are the lowest values for each of the candidate materials at temperature.
3. Classifications per NG-3220.
4. Limits on values are presented in Table 2.2.11.
5. P<sub>m</sub>, P<sub>b</sub>, Q, and P<sub>e</sub> are defined in Table 3.1.6.

**TABLE 3.1.16**

**LEVEL D: STRESS INTENSITY**

**Code:** ASME NG  
**Material:** Alloy X  
**Service Conditions:** Level D  
**Item:** Stress Intensity

<b>Temp. (Deg.F)</b>	<b>Classification and Value (ksi)</b>		
	<b>P<sub>m</sub></b>	<b>P<sub>L</sub></b>	<b>P<sub>L</sub> + P<sub>b</sub></b>
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.2	69.3	69.3
400	44.9	67.4	67.4
500	42.0	63.0	63.0
600	39.4	59.1	59.1
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.8	53.7	53.7

Notes:

1. Level D stress intensities per ASME NG-3225 and Appendix F, Paragraph F-1331.
2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S<sub>u</sub>.
3. Limits on values are presented in Table 2.2.11.
4. P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.6.

**TABLE 3.1.17**

**REFERENCE TEMPERATURES AND STRESS LIMITS  
FOR THE VARIOUS LOAD CASES**

Load Case I.D.	Material	Reference Temperature <sup>†</sup> , ° F	Stress Intensity Allowables, ksi		
			P <sub>m</sub>	P <sub>L</sub> + P <sub>b</sub>	P <sub>L</sub> + P <sub>b</sub> + Q
F1	Alloy X	725	15.4	23.1	46.2
F2	Alloy X	725	15.4	23.1	46.2
F3	Alloy X	725	36.9	55.4	NL
E1	Alloy X	<del>500</del> 500	<del>17.5</del> 18.1	<del>26.3</del> 27.2	<del>52.5</del> 54.3
E2	Alloy X	<del>500</del> 450	<del>17.5</del> 18.1	<del>26.3</del> 27.2	<del>52.5</del> 54.3
E3	Alloy X	<del>500</del> 450	<del>42.0</del> 43.4	<del>63.0</del> 5.2	NL <sup>††</sup>
E4	Alloy X	<del>500</del> 450	<del>17.5</del> 18.1	<del>26.3</del> 27.2	<del>52.5</del> 54.3
E5	Alloy X	775	36.15	54.25	NL

Note:

1. ~~1.~~ Q, P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.6.
2. Reference temperatures for Load Cases E1-E4 are for MPC shell; for MPC lid and MPC baseplate, reference temperatures are 550 deg.F and 400 deg. F, respectively (per Table 2.2.3) and stress intensity allowables should be adjusted accordingly.

<sup>†</sup> Values for reference temperatures are taken as the design temperatures (Table 2.2.3)

<sup>††</sup> NL: No specified limit in the Code

**TABLE 3.1.17 (CONTINUED)**

**REFERENCE TEMPERATURES AND STRESS LIMITS FOR THE VARIOUS LOAD CASES**

Load Case I.D.	Material	Reference Temperature, <sup>†,††</sup> ° F	Stress Intensity Allowables, ksi		
			P <sub>m</sub>	P <sub>L</sub> + P <sub>b</sub>	P <sub>L</sub> + P <sub>b</sub> + Q
O1	SA203-E	400	17.5	26.3	NL <sup>†††</sup>
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O2	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL
O3	SA203-E	400	17.5	26.3	NL
	SA350-LF3	400	17.5	26.3	NL
	SA516 Gr. 70 SA515 Gr. 70	400	17.5	26.3	NL
O4	SA203-E	400	41.2	61.7	NL
	SA350-LF3	400	38.6	58.0	NL
	SA516 Gr. 70 SA515 Gr. 70	400	39.1	58.7	NL

Note:

1. P<sub>m</sub>, P<sub>L</sub>, P<sub>b</sub>, and Q are defined in Table 3.1.6.
2. Load Cases 01 and 03 are for Normal Conditions; therefore the values listed refer to allowable stress, not allowable stress intensity

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<sup>†</sup> Values for reference temperatures are taken as the design temperatures (Table 2.2.3).

<sup>††</sup> For storage fire analysis, temperatures are defined by thermal solution

<sup>†††</sup> NL: No specified limit in the Code

**TABLE 3.1.18<sup>†</sup>**

**FRACTURE TOUGHNESS TEST REQUIREMENTS**

<b>Material</b>	<b>Test Requirement</b>	<b>Test Temperature</b>	<b>Acceptance Criterion</b>
Bolting (A193 B7)	Not required (per NF-2311(b)(13) and Note (e) to Figure NF-2311(b)-1)	-	-
Ferritic steel with nominal section thickness of 5/8" or less	Not required per NF-2311(b)(1)	-	-
SA516 Gr. 70, SA515 Gr. 70 (normalized) (thickness less than or equal to 0.75 inch)	Not required per NF-2311(b)(13) and curve D in Figure NF-2311(b)-1	-	-
SA203, SA516 Gr. 70, SA350-LF2, SA350-LF3 (greater than 0.75" thick)	Per NF-2331	See Note 1. (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2  (Also must meet ASME Section IIA requirements)
Weld material	Test per NF-2430 for welds when base metal impact testing is required.	-40 deg.F (HI-STORM) 0 deg.F (HI-TRAC) ("NF" parts) -40 deg.F (HI-TRAC) ("STH" parts)	Per NF-2330

Note:

1. Required NDT temperature = -40 deg.F for all parts in the HI-STORM 100 Overpack, -40 deg.F for HI-TRAC "STH" parts, and 0 deg.F for HI-TRAC "NF" parts.



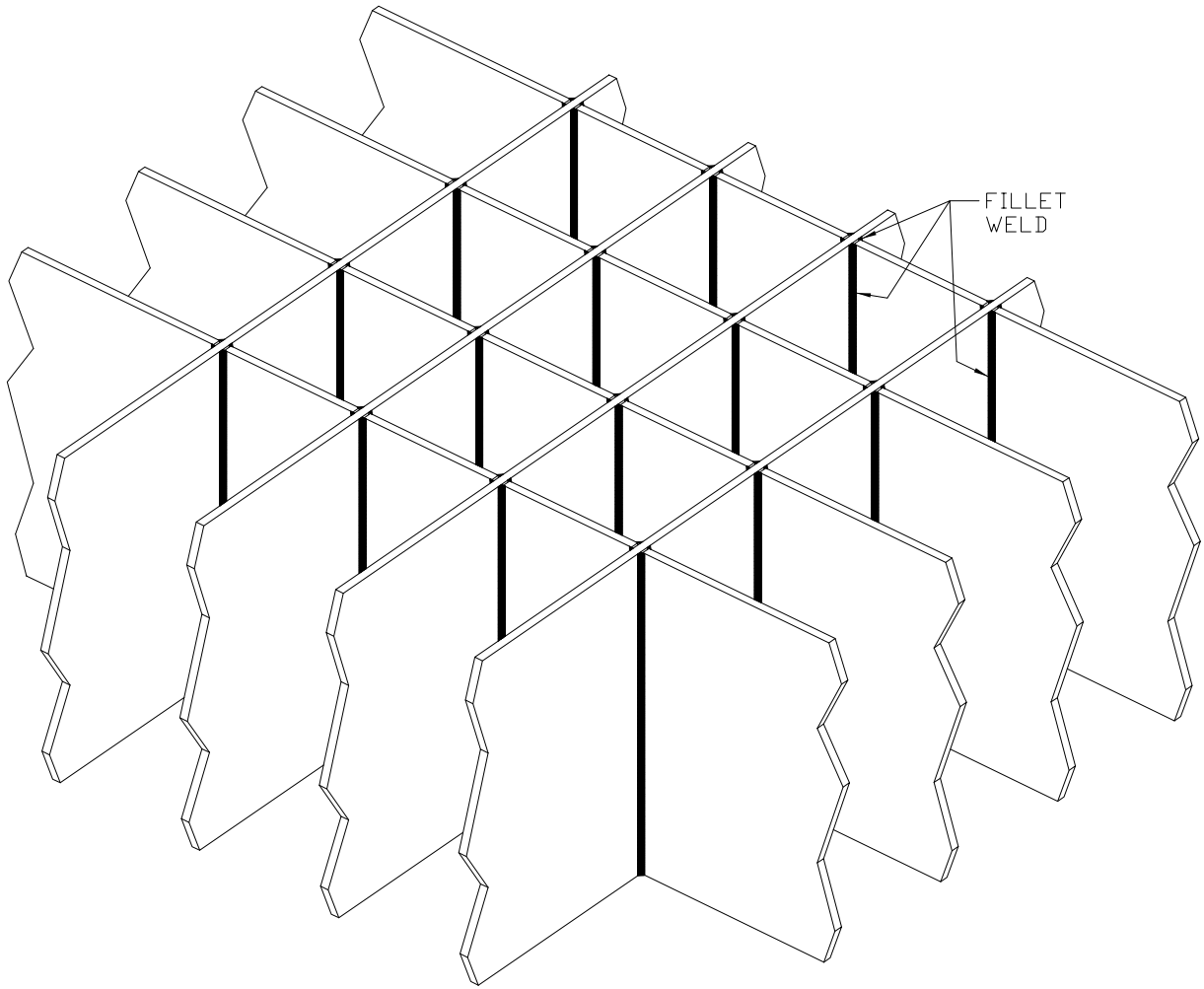


FIGURE 3.1.1; MPC-68 AND MPC-32 FUEL BASKET GEOMETRY

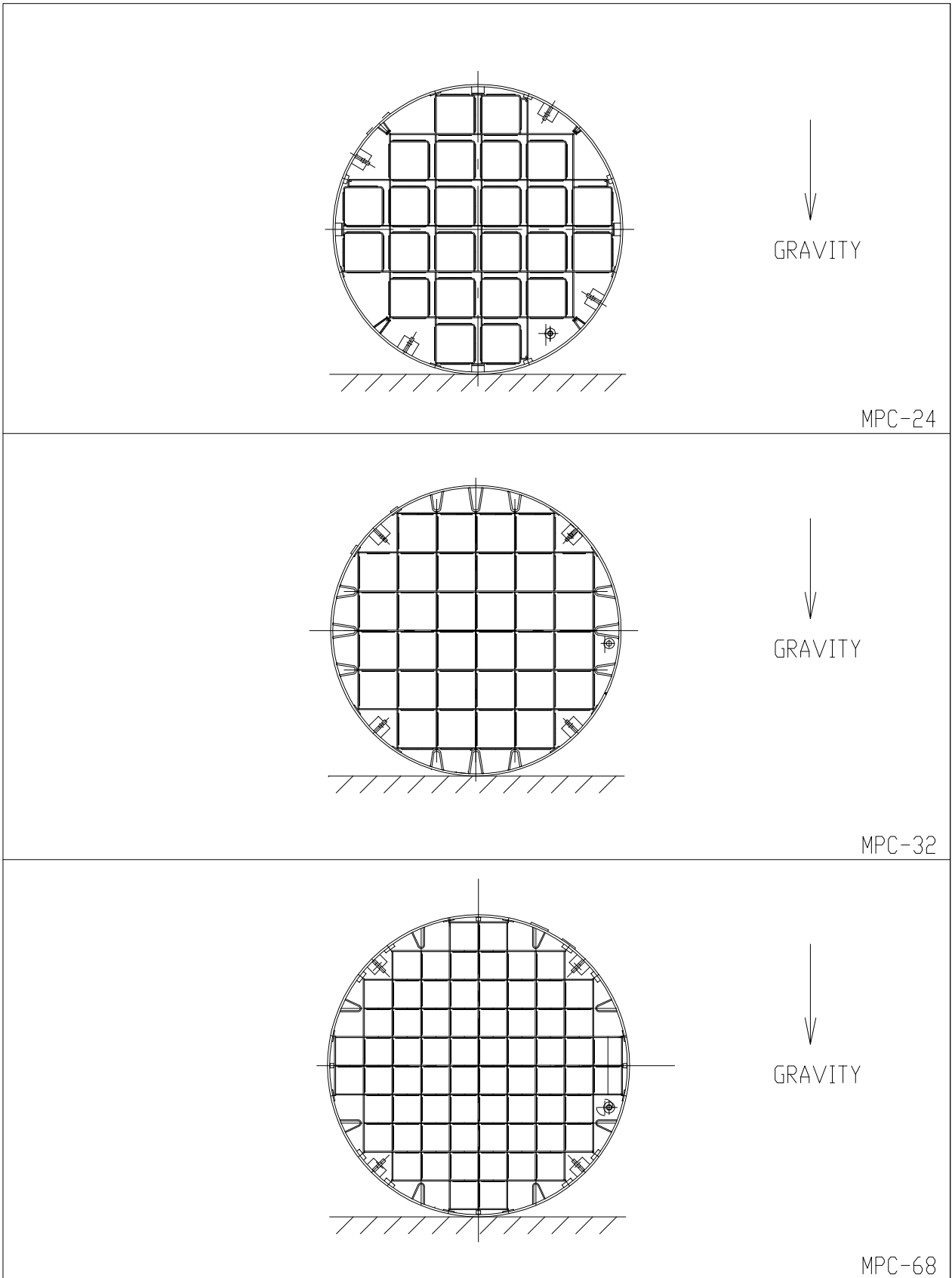
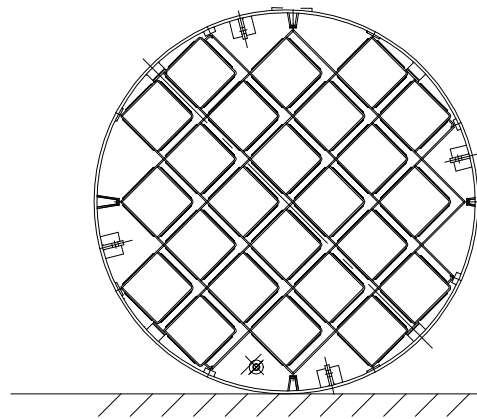
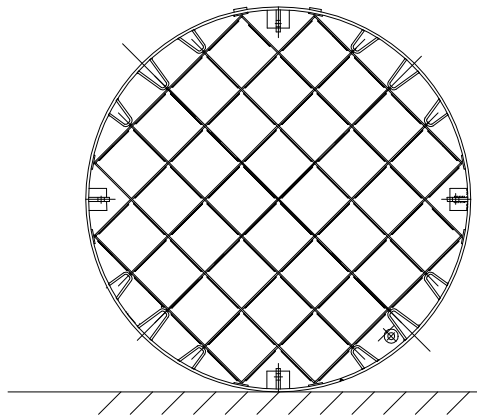


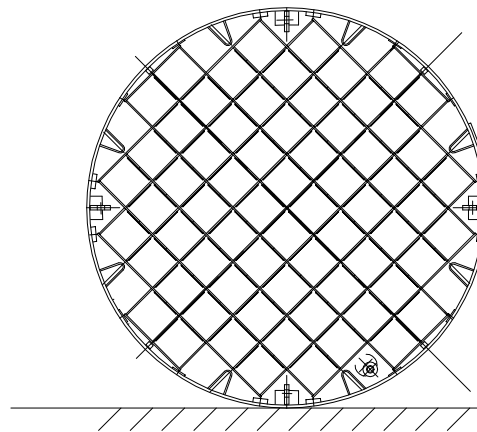
FIGURE 3.1.2; 0° DROP ORIENTATIONS FOR THE MPCs



MPC-24



MPC-32



MPC-68

*FIGURE 3.1.3; 45° DROP ORIENTATIONS FOR THE MPCs*

Tables 3.2.1 and 3.2.2 provide the calculated weights of the individual HI-STORM 100 components as well as the total system weights. The actual weights will vary within a narrow range of the calculated values due to the tolerances in metal manufacturing and fabrication permitted by the ASME Codes. Contained water mass during fuel loading is not included in this table.

The locations of the calculated centers of gravity (CGs) are presented in Table 3.2.3. All centers of gravity are located on the cask centerline since the non-axisymmetric effects of the cask system plus contents are negligible.

Table 3.2.4 provides the lift weight when the HI-TRAC transfer cask with the heaviest fully loaded MPC is being lifted from the fuel pool. The effect of buoyancy is neglected, and the weight of rigging is set at a conservative value.

In all weight tables, bounding values are also listed where necessary for use in structural calculations where their use will provide a conservative result.

**TABLE 3.2.1  
HI-STORM OVERPACK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>MPC-24</b>	
• Without SNF	42,000
• Fully loaded with SNF and Fuel Spacers	90,000
<b>MPC-32</b>	
• Without SNF	36,000
• Fully loaded with SNF and Fuel Spacers	90,000
<b>MPC-68/68F/68FF</b>	
• Without SNF	39,000
• Fully loaded with SNF and Fuel Spacers	90,000
<b>MPC-24E/EF</b>	
• Without SNF	45,000
• Fully loaded with SNF and Fuel Spacers	90,000
<b>HI-STORM 100 Overpack</b>	
• Overpack top lid	23,000
• 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
<b>HI-STORM 100S(232) Overpack</b>	
• 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

**TABLE 3.2.1 (CONTINUED)**  
**HI-STORM OVERPACK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
<b>HI-STORM 100S(243) Overpack</b> <ul style="list-style-type: none"> <li>• Overpack top lid</li> <li>• Overpack w/ lid (empty)</li> <li>• Overpack w/ fully loaded MPC-24</li> <li>• Overpack w/ fully loaded MPC-32</li> <li>• Overpack w/ fully loaded MPC-68/68F/68FF</li> <li>• Overpack w/ fully loaded MPC-24E/EF</li> </ul>	<p style="text-align: right;">25,500</p> <p style="text-align: right;">270,000</p> <p style="text-align: right;">360,000</p> <p style="text-align: right;">360,000</p> <p style="text-align: right;">360,000</p> <p style="text-align: right;">360,000</p>
<b>HI-STORM 100A Overpack</b>	Same as above

**TABLE 3.2.2  
HI-TRAC 125 TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	2,750
Pool Lid	12,500
Transfer Lid	24,500
HI-TRAC 125 w/ Top Lid and Pool Lid (water jacket filled)	143,500
HI-TRAC 125 w/ Top Lid and Transfer Lid (water jacket filled)	155,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	226,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	233,500
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF(water jacket filled)	231,000
HI-TRAC 125 w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	229,000
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	237,500
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	245,000
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	242,500
HI-TRAC 125 w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	240,500

**TABLE 3.2.2 (CONTINUED)  
HI-TRAC 100 TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	1,500
Pool Lid	8,000
Transfer Lid	17,000
HI-TRAC 100 w/ Top Lid and Pool Lid (water jacket filled)	102,000
HI-TRAC 100 w/ Top Lid and Transfer Lid (water jacket filled)	111,000
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	183,500
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	191,000
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	188,500
HI-TRAC 100 w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	186,500
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	192,000
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	199,000
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	196,500
HI-TRAC 100 w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	194,500



**TABLE 3.2.2 (CONTINUED)**  
**HI-TRAC 125D TRANSFER CASK WEIGHT DATA**

<b>Item</b>	<b>Bounding Weight (lb)</b>
Top Lid	2,750
Pool Lid	12,500
HI-TRAC 125D w/ Top Lid and Pool Lid (water jacket filled)	143,000
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	225,500
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	233,000
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	230,500
HI-TRAC 125D w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	228,500

**TABLE 3.2.3**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

Component	Height of CG Above Datum (in)
MPC-24 (empty)	109.0
MPC-32 (empty)	113.2
MPC-68/68F/68FF (empty)	111.5
MPC-24E/EF (empty)	108.9
HI-STORM 100 Overpack (empty)	116.8
HI-STORM 100S(232) Overpack (empty)	111.7
HI-STORM 100S(243) Overpack (empty)	117.4
HI-STORM 100 Overpack w/ fully loaded MPC-24	118.8
HI-STORM 100 Overpack w/ fully loaded MPC-32	118.7
HI-STORM 100 Overpack w/ fully loaded MPC-68/68F/68FF	119.0
HI-STORM 100 Overpack w/ fully loaded MPC-24E/EF	119.2
HI-STORM 100S(232) Overpack w/ fully loaded MPC-24	113.8
HI-STORM 100S(232) Overpack w/ fully loaded MPC-32	113.7
HI-STORM 100S(232) Overpack w/ fully loaded MPC-68/68F/68FF	114.0
HI-STORM 100S(232) Overpack w/ fully loaded MPC-24E/EF	114.2
HI-STORM 100S(243) Overpack w/ fully loaded MPC-24	118.1
HI-STORM 100S(243) Overpack w/ fully loaded MPC-32	117.9
HI-STORM 100S(243) Overpack w/ fully loaded MPC-68/68F/68FF	118.2
HI-STORM 100S(243) Overpack w/ fully loaded MPC-24E/EF	118.4
HI-TRAC 125 Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	93.1
HI-TRAC 125 Transfer Cask w/ Top Lid and Transfer Lid (water jacket filled)	92.3
HI-TRAC 125 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	98.0
HI-TRAC 125 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	97.9
HI-TRAC 125 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	98.3
HI-TRAC 125 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	98.6

**TABLE 3.2.3 (CONTINUED)**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

<b>Component</b>	<b>Height of CG Above Datum (in)</b>
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	99.5
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	99.5
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	99.8
HI-TRAC 125 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	100.1
HI-TRAC 100 Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	91.0
HI-TRAC 100 Transfer Cask w/ Top Lid and Transfer Lid (water jacket filled)	91.1
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	97.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	97.2
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	97.6
HI-TRAC 100 Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	98.0
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24 (water jacket filled)	100.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-32 (water jacket filled)	100.3
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	100.7
HI-TRAC 100 Transfer Cask w/ Top Lid, Transfer Lid, and fully loaded MPC-24E/EF (water jacket filled)	101.0
HI-TRAC 125D Transfer Cask w/ Top Lid and Pool Lid (water jacket filled)	92.4
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24 (water jacket filled)	97.6
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-32 (water jacket filled)	97.5

**TABLE 3.2.3 (CONTINUED)**  
**CENTERS OF GRAVITY OF HI-STORM SYSTEM CONFIGURATIONS**

Component	Height of CG Above Datum (in)
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-68/68F/68FF (water jacket filled)	97.8
HI-TRAC 125D Transfer Cask w/ Top Lid, Pool Lid, and fully loaded MPC-24E/EF (water jacket filled)	98.2

Notes:

1. The datum used for calculations involving the HI-STORM is the bottom of the overpack baseplate. The datum used for calculations involving the HI-TRAC is the bottom of the pool lid or transfer lid, as appropriate.
2. The datum used for calculations involving only the MPC is the bottom of the MPC baseplate.

**TABLE 3.2.4  
LIFT WEIGHT ABOVE POOL WITH HI-TRAC 125**

Item	Estimated Weight (lb)	Bounding Weight (lb)
HI-TRAC 125 w/ Top Lid and Pool Lid (water jacket filled)	142,976	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
HI-TRAC 125 Top Lid	-2,730 <sup>††</sup>	
Water in MPC and HI-TRAC 125 Annulus	16,570	
Water in Water Jacket	-9,757 <sup>†††</sup>	
Lift yoke	3,600	
Inflatable annulus seal	50	
<b>TOTAL</b>	240,474	245,000

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 125 includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

**TABLE 3.2.4 (CONTINUED)**  
**LIFT WEIGHT ABOVE POOL WITH HI-TRAC 100**

Item	Estimated Weight (lb)	Bounding Weight (lb)
HI-TRAC 100 w/ Top Lid and Pool Lid (water jacket filled)	100,194	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
HI-TRAC 100 Top Lid	-1,203 <sup>††</sup>	
Water in MPC and HI-TRAC 100 Annulus	16,570	
Water in Water Jacket	-7,562 <sup>†††</sup>	
Lift yoke	3,200	
Inflatable annulus seal	50	
<b>TOTAL</b>	201,014	202,000

Note: HI-TRAC transfer cask weight is without removable portion of pocket trunnion.

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 100 includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

**TABLE 3.2.4 (CONTINUED)**  
**LIFT WEIGHT ABOVE POOL WITH HI-TRAC 125D**

Item	Estimated Weight (lb)	Bounding Weight (lb)
HI-TRAC 125D w/ Top Lid and Pool Lid (water jacket filled)	142,380	
MPC-32 fully loaded with SNF and fuel spacers	89,765 <sup>†</sup>	
HI-TRAC 125D Top Lid	-2,575 <sup>††</sup>	
Water in MPC and HI-TRAC 125D Annulus	16,570	
Water in Water Jacket	-8,955 <sup>†††</sup>	
Lift yoke	3,600	
Inflatable annulus seal	50	
<b>TOTAL</b>	240,835	245,000

<sup>†</sup> Includes MPC closure ring.

<sup>††</sup> HI-TRAC top lid weight is included in transfer cask weight. However, the top lid is not installed during in-pool operations.

<sup>†††</sup> Total weight of HI-TRAC 125D includes water in water jacket. However, during removal from the fuel pool no water is in the water jacket since the water within the MPC cavity provides sufficient shielding.

Table 2.2.6 provides a comprehensive listing of materials of construction, applicable code, and ITS designation for all functional parts in the HI-STORM 100 System. This section provides the mechanical properties used in the structural evaluation. The properties include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. Values are presented for a range of temperatures which envelopes the maximum and minimum temperatures under all service conditions discussed in the preceding section where structural analysis is performed.

The materials selected for use in the MPC, HI-STORM 100 Overpack, and HI-TRAC transfer cask are presented in the Bills-of-Material in Section 1.5. In this chapter, the materials are divided into two categories, structural and nonstructural. Structural materials are materials that act as load bearing members and are, therefore, significant in the stress evaluations. Materials that do not support mechanical loads are considered nonstructural. For example, the HI-TRAC inner shell is a structural material, while the lead between the inner and outer shell is a nonstructural material. For nonstructural materials, the only property that is used in the structural analysis is weight density. In local deformation analysis, however, such as the study of penetration from a tornado-borne missile, the properties of lead in HI-TRAC and plain concrete in HI-STORM 100, are included.

### 3.3.1 Structural Materials

#### 3.3.1.1 Alloy X

A hypothetical material termed Alloy X is defined for all MPC structural components. The material properties of Alloy X are the least favorable values from the set of candidate alloys. The purpose of a least favorable material definition is to ensure that all structural analyses are conservative, regardless of the actual MPC material. For example, when evaluating the stresses in the MPC, it is conservative to work with the minimum values for yield strength and ultimate strength. This guarantees that the material used for fabrication of the MPC will be of equal or greater strength than the hypothetical material used in the analysis. In the structural evaluation, the only property for which it is not always conservative to use the set of minimum values is the coefficient of thermal expansion. Two sets of values for the coefficient of thermal expansion are specified, a minimum set and a maximum set. For each analysis, the set of coefficients, minimum or maximum that causes the more severe load on the cask system is used.

Table 3.3.1 lists the numerical values for the material properties of Alloy X versus temperature. These values, taken from the ASME Code, Section II, Part D [3.3.1], are used in all structural analyses. The maximum temperatures in some MPC components may exceed the allowable limits of temperature during short time duration loading operations, off-normal transfer operations, or storage accident events. However, no maximum temperature for Alloy X used at or within the confinement boundary exceeds 1000°F. As shown in ASME Code Case N-47-33 (Class 1 Components in Elevated Temperature Service, 1995 Code Cases, Nuclear Components), the strength properties of austenitic stainless steels do not change due to exposure to 1000°F temperature for up to 10,000 hours. Therefore, there is no significant effect on



mechanical properties of the confinement or basket material during the short time duration loading. A further description of Alloy X, including the materials from which it is derived, is provided in Appendix 1.A.

Two properties of Alloy X that are not included in Table 3.3.1 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses, regardless of temperature. The values used are shown in the table below.

<b>PROPERTY</b>	<b>VALUE</b>
Weight Density (lb/in <sup>3</sup> )	0.290
Poisson's Ratio	0.30

### 3.3.1.2 Carbon Steel, Low-Alloy and Nickel Alloy Steel

The carbon steels in the HI-STORM 100 System are SA516 Grade 70 and SA515 Grade 70. The nickel alloy and low alloy steels are SA203-E and SA350-LF3, respectively. These steels are not constituents of Alloy X. The material properties of SA516 Grade 70 and SA515 Grade 70 are shown in Tables 3.3.2. The material properties of SA203-E and SA350-LF3 are given in Table 3.3.3.

Two properties of these steels that are not included in Tables 3.3.2 and 3.3.3 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses. The values used are shown in the table below.

<b>PROPERTY</b>	<b>VALUE</b>
Weight Density (lb/in <sup>3</sup> )	0.283
Poisson's Ratio	0.30

### 3.3.1.3 Bolting Materials

Material properties of the bolting materials used in the HI-STORM 100 System and HI-TRAC lifting trunnions are given in Table 3.3.4. The properties of representative anchor studs used to fasten HI-STORM 100A are listed in Table 1.2.7.

### 3.3.1.4 Weld Material

All weld materials utilized in the welding of the Code components comply with the provisions of the appropriate ASME subsection (e.g., Subsection NB for the MPC enclosure vessel) and Section IX. All non-code welds will be made using weld procedures that meet Section IX of the ASME Code. The minimum tensile strength of the weld wire and filler material (where applicable) will be equal to or greater than the tensile strength of the base metal listed in the ASME Code.

### 3.3.2 Nonstructural Materials

#### 3.3.2.1 Solid Neutron Shield

The solid neutron shielding material in the HI-TRAC top lid and transfer lid doors is not considered as a structural member of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analyses except where such omission would be non-conservative. The only material property of the solid neutron shield that is important to the structural evaluation is weight density ( $1.63\text{g/cm}^3$ ).

#### 3.3.2.2 Boral<sup>TM</sup>-Solid Neutron Absorber

~~Boral~~The fuel basket solid neutron absorber is not a structural members of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analyses. The only material property of ~~Boral~~the solid neutron absorber that is important to the structural evaluation is weight density. As the MPC fuel baskets can be constructed with ~~Boral~~neutron absorber panels of variable areal density, the weight that produces the most severe cask load is assumed in each analysis (density  $2.644\text{ g/cm}^3$ ).

#### 3.3.2.3 Concrete

The primary function of the plain concrete in the HI-STORM storage overpack is shielding. Concrete in the HI-STORM 100 Overpack is not considered as a structural member, except to withstand compressive, bearing, and penetrant loads. While concrete is not considered a structural member, its mechanical behavior must be quantified to determine the stresses in the structural members (steel shells surrounding it) under accident conditions. Table 3.3.5 provides the concrete mechanical properties. Allowable, bearing strength in concrete for normal loading conditions is calculated in accordance with ACI 318.1 [3.3.2]. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. In addition, although the concrete is not reinforced (since the absence of reinforcement does not degrade the compressive strength), the requirements of ACI-349 [3.3.3] are imposed to insure the suitability of the concrete mix. Appendix 1.D provides additional information on the requirements on plain concrete for use in HI-STORM 100 storage overpack.

#### 3.3.2.4 Lead

Lead is not considered as a structural member of the HI-STORM 100 System. Its load carrying capacity is neglected in all structural analysis, except in the analysis of a tornado missile strike where it acts as a missile barrier. Applicable mechanical properties of lead are provided in Table 3.3.5.

#### 3.3.2.5 Aluminum Heat Conduction Elements

Optional aluminum heat conduction elements may be located between the fuel basket and MPC vessel. They are optional thin flexible elements whose sole function is to transmit heat as described

in Chapter 4. They are not credited with any structural load capacity and are shaped to provide negligible resistance to basket thermal expansion. The total weight of the aluminum inserts is less than 1,000 lb. per MPC.

**TABLE 3.3.1  
ALLOY X MATERIAL PROPERTIES**

Temp. (Deg. F)	Alloy X				
	S <sub>y</sub>	S <sub>u</sub>	α <sub>min</sub>	α <sub>max</sub>	E
-40	30.0	75.0	8.54	8.55	28.82
100	30.0	75.0	8.54	8.55	28.14
150	27.5	73.0	8.64	8.67	27.87
200	25.0	71.0	8.76	8.79	27.6
250	23.75	68.5	8.88	8.9	27.3
300	22.5	66.0	8.97	9.0	27.0
350	21.6	65.2	9.10	9.11	26.75
400	20.7	64.4	9.19	9.21	26.5
450	20.05	64.0	9.28	9.32	26.15
500	19.4	63.5	9.37	9.42	25.8
550	18.8	63.3	9.45	9.50	25.55
600	18.2	63.1	9.53	9.6	25.3
650	17.8	62.8	9.61	9.69	25.05
700	17.3	62.5	9.69	9.76	24.8
750	16.9	62.2	9.76	9.81	24.45
800	16.6	61.7	9.82	9.90	24.1

Definitions:

- S<sub>y</sub> = Yield Stress (ksi)
- α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)
- S<sub>u</sub> = Ultimate Stress (ksi)
- E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α<sub>min</sub> and α<sub>max</sub> values is Table TE-1 of [3.3.1].
4. Source for E values is material group G in Table TM-1 of [3.3.1].

**TABLE 3.3.2  
SA516 AND SA515, GRADE 70 MATERIAL PROPERTIES**

Temp. (Deg.F)	SA516 and SA515, Grade 70			
	S <sub>y</sub>	S <sub>u</sub>	α	E
-40	38.0	70.0	---	29.95
100	38.0	70.0	5.53 (5.73)	29.34
150	36.3	70.0	5.71 (5.91)	29.1
200	34.6	70.0	5.89 (6.09)	28.8
250	34.15	70.0	6.09 (6.27)	28.6
300	33.7	70.0	6.26 (6.43)	28.3
350	33.15	70.0	6.43 (6.59)	28.0
400	32.6	70.0	6.61 (6.74)	27.7
450	31.65	70.0	6.77 (6.89)	27.5
500	30.7	70.0	6.91 (7.06)	27.3
550	29.4	70.0	7.06 (7.18)	27.0
600	28.1	70.0	7.17 (7.28)	26.7
650	27.6	70.0	7.30 (7.40)	26.1
700	27.4	70.0	7.41 (7.51)	25.5
750	26.5	69.3	7.50 (7.61)	24.85

Definitions:

S<sub>y</sub> = Yield Stress (ksi)

α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress (ksi)

E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α values is material group C in Table TE-1 of [3.3.1].
4. Source for E values is "Carbon steels with C less than or equal to 0.30%" in Table TM-1 of [3.3.1].
5. Values for SA515 are given in parentheses where different from SA516.

**TABLE 3.3.3  
SA350-LF3 AND SA203-E MATERIAL PROPERTIES**

Temp. (Deg.F)	SA350-LF3 and LF2			SA350-LF3/SA203-E		SA203-E		
	S <sub>m</sub>	S <sub>y</sub>	S <sub>u</sub>	E	α	S <sub>m</sub>	S <sub>y</sub>	S <sub>u</sub>
-20	23.3	37.5 (36.0)	70.0	28.2	---	23.3	40.0	70.0
100	23.3	37.5 (36.0)	70.0	27.6	6.27	23.3	40.0	70.0
200	22.8 (21.9)	34.2 (32.9)	68.5 (70.0)	27.1	6.54	23.3	36.5	70.0
300	22.2 (21.3)	33.2 (31.9)	66.7 (70.0)	26.7	6.78	23.3	35.4	70.0
400	21.5 (20.6)	32.2 (30.9)	64.6 (70.0)	26.1	6.98	22.9	34.3	68.8
500	20.2 (19.4)	30.3 (29.2)	60.7 (70.0)	25.7	7.16	21.6	32.4	64.9
600	18.5 (17.8)	-(26.6)	-(70.0)	-	-	-	-	-
700	16.8 (17.3)	-(26.0)	-(70.0)	-	-	-	-	-

Definitions:

- S<sub>m</sub> = Design Stress Intensity (ksi)
- S<sub>y</sub> = Yield Stress (ksi)
- S<sub>u</sub> = Ultimate Stress (ksi)
- α = Coefficient of Thermal Expansion (in./in. per degree F x 10<sup>-6</sup>)
- E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>m</sub> values is ASME Code.
2. Source for S<sub>y</sub> values is ASME Code.
3. Source for S<sub>u</sub> values is ratioing S<sub>m</sub> values.
4. Source for α values is material group E in Table TE-1 of [3.3.1].
5. Source for E values is material group B in Table TM-1 of [3.3.1].
6. Values for LF2 are given in parentheses where different from LF3

**TABLE 3.3.4  
BOLTING MATERIAL PROPERTIES**

Temp. (Deg.F)	SB637-N07718				
	S <sub>y</sub>	S <sub>u</sub>	E	α	S <sub>m</sub>
-100	150.0	185.0	29.9	---	50.0
-20	150.0	185.0	---	---	50.0
70	150.0	185.0	29.0	7.05	50.0
100	150.0	185.0	---	7.08	50.0
200	144.0	177.6	28.3	7.22	48.0
300	140.7	173.5	27.8	7.33	46.9
400	138.3	170.6	27.6	7.45	46.1
500	136.8	168.7	27.1	7.57	45.6
600	135.3	166.9	26.8	7.67	45.1
SA193 Grade B7 (2.5 to 4 inches diameter)					
Temp. (Deg. F)	S <sub>y</sub>	S <sub>u</sub>	E	α	-
100	95.0	115.00	-	5.73	-
200	88.5	107.13	-	6.09	-
300	85.1	103.02	-	6.43	-
400	82.3	99.63	-	5.9	-

Definitions:

S<sub>m</sub> = Design stress intensity (ksi)

S<sub>y</sub> = Yield Stress (ksi)

α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress (ksi)

E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>m</sub> values is Table 4 of [3.3.1].
2. Source for S<sub>y</sub> values is ratioing design stress intensity values.
3. Source for S<sub>u</sub> values is ratioing design stress intensity values.
4. Source for α values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
5. Source for E values is Table TM-1 of [3.3.1].
6. Source for S<sub>y</sub> values for SA193 bolts is Table Y-1 of [3.3.1]; source for S<sub>u</sub> is by ratioing S<sub>y</sub>.

**TABLE 3.3.4 (CONTINUED)  
BOLTING MATERIAL PROPERTIES**

SA193 Grade B7 (less than 2.5 inch diameter)					
Temp. (Deg.F)	S <sub>y</sub>	S <sub>u</sub>	E	α	-
100	105.0	125.00	-	5.73	-
200	98.0	116.67	-	6.09	-
300	94.1	112.02	-	6.43	
400	91.5	108.93	-	6.74	-
Temp. (Deg.F)	SA705-630/SA564-630 (Age Hardened at 1075 degrees F)				
	S <sub>y</sub>	S <sub>u</sub>	E	α	S <sub>m</sub>
200	115.6	145.0	28.5	5.9	---
300	110.7	145.0	27.9	5.9	---
400	106.9	145.0	27.3	5.91	---
SA705-630/SA564-630 (Age Hardened at 1150 degrees F)					
200	97.1	135.0	28.5	5.9	---
300	93.0	135.0	27.9	5.9	---

Definitions:

S<sub>m</sub> = Design stress intensity (ksi)

S<sub>y</sub> = Yield Stress (ksi)

α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)

S<sub>u</sub> = Ultimate Stress (ksi)

E = Young's Modulus (psi x 10<sup>6</sup>)

Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
4. Source for E values is Table TM-1 of [3.3.1].



**TABLE 3.3.5  
CONCRETE AND LEAD MECHANICAL PROPERTIES**

PROPERTY	VALUE					
<b>CONCRETE:</b>						
Compressive Strength (psi)	See Table 1.D.1					
Nominal Density (lb/ft <sup>3</sup> )	See Table 1.D.1					
Allowable Bearing Stress (psi)	1,823 <sup>†</sup>					
Allowable Axial Compression (psi)	1,266 <sup>†</sup>					
Allowable Flexure, extreme fiber tension (psi)	187 <sup>†,††</sup>					
Allowable Flexure, extreme fiber compression (psi)	2,145 <sup>†</sup>					
Mean Coefficient of Thermal Expansion (in/in/deg.F)	5.5E-06					
Modulus of Elasticity (psi)	57,000 (compressive strength (psi)) <sup>1/2</sup>					
<b>LEAD:</b>	-40°F	-20°F	70°F	200°F	300°F	600°F
Yield Strength (psi)	700	680	640	490	380	20
Modulus of Elasticity (ksi)	2.4E+3	2.4E+3	2.3E+3	2.0E+3	1.9E+3	1.5E+3
Coefficient of Thermal Expansion (in/in/deg.F)	15.6E-6	15.7E-6	16.1E-6	16.6E-6	17.2E-6	20.2E-6
Poisson's Ratio	0.40					
Density (lb/cubic ft.)	708					

Notes:

- Concrete allowable stress values based on ACI 318.1.
- Lead properties are from [3.3.5].

<sup>†</sup> Values listed correspond to concrete compressive stress = 4,000 psi

<sup>††</sup> No credit for tensile strength of concrete is taken in the calculations

## 3.4 GENERAL STANDARDS FOR CASKS

### 3.4.1 Chemical and Galvanic Reactions

In this section, it is shown that there is no credible mechanism for significant chemical or galvanic reactions in the HI-STORM 100 System during long-term storage operations (including HI-STORM 100S and HI-STORM 100A).

The MPC, which is filled with helium, provides a nonaqueous and inert environment. Insofar as corrosion is a long-term time-dependent phenomenon, the inert gas environment in the MPC precludes the incidence of corrosion during storage on the ISFSI. Furthermore, the only dissimilar material groups in the MPC are: (1) *the neutron absorber material Boral<sup>TM</sup>* and stainless steel and (2) aluminum and stainless steel. *Neutron absorber materials Boral* and stainless steels have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from *neutron absorber materials Boral* and stainless steel materials, with geometries similar to the MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This experience provides a sound basis to conclude that corrosion will not occur in these materials. Additionally, the aluminum conduction inserts and stainless steel basket are very close on the galvanic series chart. Aluminum, like other metals of its genre (e.g., titanium and magnesium) rapidly passivates in an aqueous environment, leading to a thin ceramic ( $Al_2O_3$ ) barrier, which renders the material essentially inert and corrosion-free over long periods of application. The physical properties of the material, e.g., thermal expansion coefficient, diffusivity, and thermal conductivity, are essentially unaltered by the exposure of the aluminum metal stock to an aqueous environment.

~~In order to minimize the incidence of aluminum water reaction inside the MPC during fuel loading operation (when the MPC is flooded with pool water) all aluminum surfaces are pre-passivated or anodized before installation of Boral or optional aluminum heat conduction inserts in the MPC. Because the aluminum-water reaction cannot be completely eliminated by pre-passivation and the aluminum material in the MPC will be under varying hydrostatic pressure levels (up to approximately 40 feet of water pressure during fuel loading or unloading in the spent fuel pool, and up to approximately 15 feet during lid welding or cutting), continued generation of limited quantities of hydrogen is possible. Pre-passivation has been shown by analysis to preclude the accumulation of combustible quantities of gas under the MPC lid during welding or cutting. However, as a defense in depth measure to preclude the potential for ignition during the conduct of these activities, the operating procedures in Chapter 8 include a requirement for periodic combustible gas monitoring and recommended actions to evacuate, or purge the space beneath the MPC lid with an inert gas prior to and during lid welding and cutting activities.~~

*The aluminum in the optional heat conduction elements will quickly passivate in air and in water to form a protective oxide layer that prevents any significant hydrogen production during MPC cask loading and unloading operations. The aluminum in the neutron absorber material (i.e., Boral), particularly in the core area, will also react with the water to generate hydrogen gas. The exact rate of generation and total amount of hydrogen generated is a function of a number of variables (see Section 1.2.1.3.1.1) and cannot be predicted with any certainty. Therefore, to preclude the potential for hydrogen ignition during lid welding or cutting, the operating procedures in Chapter 8 require monitoring for combustible gas and either exhausting or purging the space beneath the*

*MPC lid with an inert gas during these activities. Once the MPC cavity is drained, dried, and backfilled with helium, the source of the hydrogen gas (the aluminum-water reaction) is eliminated.*

The HI-STORM 100 storage overpack and the HI-TRAC transfer cask each combine low alloy and nickel alloy steels, carbon steels, neutron and gamma shielding materials, and bolting materials. All of these materials have a long history of nongalvanic behavior within close proximity of each other. The internal and external steel surfaces of each of the storage overpacks are sandblasted and coated to preclude surface oxidation. The HI-TRAC coating does not chemically react with borated water. Therefore, chemical or galvanic reactions involving the storage overpack materials are highly unlikely and are not expected.

In accordance with NRC Bulletin 96-04 [3.4.7], a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-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 provided~~ 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 requires that under the "apparent dead load",  $D^*$ , the maximum primary stress in the trunnion be less than 10% of the trunnion material ultimate strength and less than 1/6th of the trunnion material yield strength. 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.

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. ~~Appendix 3.E contains the lifting trunnion stress analysis for the HI-TRAC 125. Figures within that appendix provide details to support the analysis. The lifting analysis demonstrates~~ It is demonstrated in Appendix 3.E that the stresses in the trunnions, computed using the conservative methodology described previously, comply with NUREG-0612 provisions.

Specifically, the following results are obtained:

<b>HI-TRAC 125 Lifting Trunnions<sup>†</sup></b>		
	<b>Value (ksi)</b>	<b>Safety Factor</b>
Bending stress	16.98	1.07
Shear stress	7.23	1.5

<sup>†</sup> The lifted load is 245,000 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 6 on yield strength and 10 on tensile strength. The results above are also valid for the HI-TRAC 125D since the dimensions used as input in Appendix 3-E, as well as the bounding load, are common to both the HI-TRAC 125 and 125D transfer casks.

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

~~Appendix 3-E contains calculations that analyze~~ 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:

<b>125 Ton HI-TRAC Lifting Trunnion Block (Region A Evaluation)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Trunnion Block Bearing Stress	5.94	11.4	1.92
Trunnion Block Thread Shear Stress	5.19	6.84	1.32
Weld Shear Stress (3D*)	4.40 <sup>†</sup>	11.4	2.59

<sup>†</sup> No quality factor has been applied to the weld group. (Subsection NF or NUREG-0612 do not apply penalty factors to the structural welds).

### 3.4.3.3 125 Ton HI-TRAC Lifting - Structure near Trunnion (Region B/Region A)

~~Appendix 3.AE contains results of a finite element analysis of the region in the HI-TRAC 125 structure adjacent to the lifting trunnions. Appendix 3.AE shows that the primary stresses in the HI-TRAC 125 structure comply with the Level A stress limits for Subsection NF structures.~~

A three-dimensional elastic model of the HI-TRAC 125 metal components is analyzed using the ANSYS finite element code. ~~Figure 3.AE.1 shows details of the one-quarter symmetry model using a color coding to identify the various modeled parts.~~ The structural model includes, in addition to the trunnion and the trunnion block, a portion of the inner and outer HI-TRAC shells and the HI-TRAC top flange. ~~In Appendix 3.AE, stress results over the characteristic interface section are summarized and compared with allowable strength limits per ASME Section III, Subsection NF, and per Regulatory Guide 3.61. The results show that the primary stresses in the HI-TRAC 125 structure comply with the Level A stress limits for Subsection NF structures.~~

The results from the analysis ~~in Appendix 3.AE~~ are summarized below:

<b>HI-TRAC 125 Trunnion Region (Regions A and B)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Membrane Stress	6.19	17.5	2.83
Membrane plus Bending Stress	8.19	26.25	3.2
Membrane Stress (3D*)	18.6	34.6	1.86

The results above are also valid for the HI-TRAC 125D since the dimensions and the configuration of the inner shell, outer shell, top flange, and the trunnion block are the same in both the HI-TRAC 125 and 125D transfer casks.

### 3.4.3.4 100 Ton HI-TRAC Lifting Analysis

The lifting trunnions and the trunnion blocks for the 100 Ton HI-TRAC are identical to the trunnions analyzed ~~in Appendices 3.E and 3.AE~~ for the 125 Ton HI-TRAC. However, the outer shell geometry (outer diameter) is different. A calculation performed in the spirit of strength-of-materials provides justification that, despite the difference in local structure at the attachment points, the stresses in the body of the HI-TRAC 100 Ton unit meet the allowables set forth in Subsection 3.1.2.2.

Figure 3.4.10 illustrates the differences in geometry, loads, and trunnion moment arms between the body of the 125-Ton HI-TRAC and the body of the 100-Ton HI-TRAC. It is reasonable to assume that the level of stress in the 100 Ton HI-TRAC body, in the immediate vicinity of the interface (Section X-X in Figure 3.4.10), is proportional to the applied force and the bending moment applied. ~~In the figure what follows,~~ the subscripts 1 and 0 refer to 100 Ton and 125 Ton casks, respectively.

Figure 3.4.10 shows the location of the area centroid (with respect to the outer surface) and the loads and moment arms associated with each construction. Conservatively, neglecting all other interfaces between the top of the trunnion block and the top flange and between the sides of the trunnion block and the shells, equilibrium is maintained by developing a force and a moment in the section comprised of the two shell segments interfacing with the base of the trunnion block.

The most limiting stress state is in the outer shell at the trunnion block base interface. The stress level in the outer shell at Section X-X is proportional to  $P/A + Mc/I$ . Evaluating the stress for a unit width of section permits an estimate of the stress state in the HI-TRAC 100 outer shell if the corresponding stress state in the HI-TRAC 125 is known (the only changes are the applied load, the moment arm and the geometry). Using the geometry shown in Figure 3.4.10 gives the result as:

$$\text{Stress (HI-TRAC 100 outer shell)} = 1.236 \times \text{Stress (HI-TRAC 125 outer shell)}$$

The tabular results in the previous subsection can be adjusted accordingly and are reported below:

<b>100 Ton HI-TRAC Near Trunnion (Region A and Region B)</b>	
<b>Item</b>	<b>Safety Factor</b>
Membrane Stress	2.29
Membrane plus Bending Stress	2.59
Membrane Stress (3D*)	1.50

### 3.4.3.5 HI-STORM 100 Lifting Analyses

There are two vertical lifting scenarios for the HI-STORM 100 storage overpack carrying a fully loaded MPC. Figure 3.4.17 shows a schematic of these lifting scenarios. Both lifting scenarios are examined in Appendix 3.D using finite element models that focus on the local regions near the lift points. The analysis in Appendix 3.D is based on the geometry of the HI-STORM 100; the alterations to the lid and to the length of the overpack barrel to configure the HI-STORM 100S have no effect on the conclusions reached in the area of the baseplate. Therefore, there is no separate analysis for the analysis of the baseplate, inboard of the inner shell, for the HI-STORM 100S as the results are identical to or bounded by the results presented here documented in Appendix 3.D. Since the upper portion of the HI-STORM 100S, the HI-STORM 100S lid, and the radial ribs and anchor block have a different configuration than the HI-STORM 100, separate calculations have been performed for these areas of the HI-STORM 100S.

Scenario #1 considers a "bottom lift" where the fully loaded HI-STORM 100 storage overpack is lifted vertically by four synchronized hydraulic jacks each positioned at one of the four inlet air vents. This lift allows for installation and removal of "air pads" which may be used for horizontal positioning of HI-STORM 100 at the ISFSI pad.



Scenario #2, labeled the "top lift scenario" considers the lifting of a fully loaded HI-STORM 100 vertically through the four lifting lugs located at the top end.

No structural credit is assumed for the HI-STORM concrete in either of the two lifting scenarios except as a vehicle to transfer compressive loads.

For the bottom lift, a three-dimensional one-quarter symmetry finite element model of the bottom region of the HI-STORM 100 storage overpack is constructed. The model includes the inner shell, the outer shell, the baseplate, the inlet vent side and top plates, and the radial plates connecting the inner and outer shells. ~~Further details of the model are provided in Appendix 3.D. The key results are contained in Figure 3.D.3 that shows the stress intensity distribution on the HI-STORM 100 storage overpack.~~

For the analysis of the "top lift" scenario, a three-dimensional 1/8-symmetry finite element model of the top segment of HI-STORM 100 storage overpack is constructed. The metal HI-STORM 100 material is modeled (shells, radial plates, lifting block, ribs, vent plates, etc.) using shell or solid elements. ~~Color-coded views of the model are given in Figure 3.D.2.~~ Lumped weights are used to ensure that portions of the structure not modeled are, in fact, properly represented as part of a lifted load. The model is supported vertically at the lifting lug. *The results are reported in tabular form at the end of this subsection.*

~~Figures 3.D.4(a) through 3.D.4(e) and Figure 3.D.5(a) through 3.D.5(e) show the stress intensity results under the lifted load and in the baseplate region, respectively.~~

To provide an alternate calculation to demonstrate that the bolt anchor blocks are adequate, we compute the average normal stress in the net metal area of the block under three times the lifted load. Further conservatism is introduced by including an additional 15% for dynamic amplification, i.e., the total load is equal to 3D\*.

The average normal load in one bolt anchor block is

$$\text{Load} = 3 \times 1.15 \times 360,000 \text{ lb.} / 4 = 310,500 \text{ lb.} \quad (\text{Weight comes from Table 3.2.1})$$

The net area of the bolt anchor block is

$$\text{Area} = 5'' \times 5'' - (3.14159/4)/4 \times (3.25'' \times 3.25'') = 16.70 \text{ sq. inch} \quad (\text{Dimensions from BM-1575})$$

Therefore, the safety factor (yield strength at 350 degrees F/calculated stress from Table 3.3.3) is

$$\text{SF} = 32,700 \text{ psi} / (\text{Load}/\text{Area}) = 1.76$$

Appendix 3.D also examines the shear stress in the threads of the lifting block *is also examined*. This analysis considers a cylindrical area of material under an axial load resisting the load by shearing action. The diameter of the area is the basic pitch diameter of the threads, and the length of the cylinder is the thread engagement length.

The analysis Appendix 3.D also examines the capacity of major welds in the load path and the compression capacity of the pedestal shield and pedestal shield shell.

The table below summarizes key results obtained from the analyses *described above* reported in detail in Appendix 3.D for the HI-STORM 100.

<b>HI-STORM 100 Top and Bottom Lifting Analyses<sup>†‡</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Primary Membrane plus Bending - Bottom Lift - Inlet Vent Plates - Region B	8.0	26.3	3.28
Primary Membrane - Top Lift - Radial Rib Under Lifting Block - Region B	6.67	17.5	2.63
Primary Membrane plus Bending – Top Lift - Baseplate – Region B	7.0	26.3	3.75
Primary Membrane Region A (3D*)	19.97	33.15	1.66
Primary Membrane plus Bending Region A (3D*)	24.02	33.15	1.38
Lifting Block Threads - Top Lift –Region A (3D*)	10.67	19.62	1.84
Lifting Stud - Top Lift –Region A (3D*)	43.733	108.8	2.49
Welds – Anchor Block-to-Radial Rib Region B	5.74	19.695	3.43
Welds – Anchor Block-to-Radial Rib Region A (3D*)	17.21	19.62	1.14
Welds – Radial Rib-to-Inner and Outer Shells Region B	5.83	21.00	3.60
Welds – Radial Rib-to-Inner and Outer Shells Region A (3D*)	17.49	19.89	1.13
Weld – Baseplate-to Inner Shell Region A (3D*)	1.59	19.89	12.48
Weld – Baseplate-to-Inlet Vent Region A (3D*)	14.89	19.89	1.33
Pedestal Shield Concrete (3D*)	0.096	1.266	13.19
Pedestal Shell (3D*)	3.269	33.15	10.14

<sup>†</sup> Regions A and B are defined at beginning of Subsection 3.4.3

<sup>‡</sup> The lifted load is 360000 lb. and an inertia amplification of 15% is included.

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100 storage overpack under either the top lift or the bottom lift scenario. All factors of safety are greater than 1.0 using criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports and from USNRC Regulatory Guide 3.61.

Similar calculations have been performed for the HI-STORM 100S where differences in configuration warrant. The results are summarized in the table below:

<b>HI-STORM 100S Top and Bottom Lifting Analyses<sup>†‡</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Primary Membrane plus Bending - Bottom Lift - Inlet Vent Plates - Region A (3D*)	9.824	33.15	3.374
Lifting Block Threads - Top Lift –Region A (3D*)	5.540	18.840	3.40
Lifting Stud - Top Lift –Region A (3D*)	49.199	83.7	1.70
Welds – Anchor Block-to-Radial Rib Region B	5.483	21.0	3.83
Welds – Anchor Block-to-Radial Rib Region A (3D*)	16.469	18.84	1.144
Welds – Radial Rib-to-Inner and Outer Shells Region B	5.56	21.00	3.77
Welds – Radial Rib-to-Inner and Outer Shells Region A (3D*)	16.69	19.89	1.19
Weld – Baseplate-to Inner Shell Region A (3D*)	1,592	19.89	12.49
Weld – Baseplate-to-Inlet Vent Region A (3D*)	8.982	19.89	2.214
Radial Rib Membrane Stress – Bottom Lift Region A (3D*)	10.58	33.15	3.132
Pedestal Shield Concrete (3D*)	0.095	1.535	16.17
Pedestal Shell (3D*)	3.235	33.15	10.24

<sup>†</sup> Regions A and B are defined at beginning of Subsection 3.4.3

<sup>‡</sup> The lifted load is 405,000 lb. and an inertia amplification of 15% is included. The increased weight (over the longer HI-STORM 100) comes from conservatively assuming an increase in concrete weight density in the HI-STORM 100S overpack and lid to provide additional safety margin.

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100 and HI-STORM 100S storage overpacks under either the top lift or the bottom lift scenario. All factors of safety are greater than 1.0 using criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports and from USNRC Regulatory Guide 3.61.

### 3.4.3.6 MPC Lifting Analysis

The MPC lifting analyses are found in the HI-STAR 100 FSAR (Docket-72-1008). Some results of the analyses in that document (Appendices 3.K, 3.E, 3.I and 3.Y Docket-72-1008) are summarized here for completeness.

<b>Summary of MPC Lifting Analyses</b>			
<b>Item</b>	<b>Thread Engagement Safety Factor (NUREG-0612)</b>	<b>Region A Safety Factor</b>	<b>Region B Safety Factor<sup>†</sup></b>
MPC	1.08	1.09	1.56

<sup>†</sup> The factor reported here is for the MPC baseplate considered under a load equal to 3D\*.

### 3.4.3.7 Miscellaneous Lid Lifting Analyses

~~Appendix 3.AC contains analyses of lifting attachments for various lid lifting operations.~~

The HI-STORM 100 lid lifting analysis is performed to ensure that the threaded connections provided in the lid are adequately sized. The lifting analysis of the top lid is based on a vertical orientation of loading from an attached lifting device. The top lid of the HI-STORM 100 storage overpack is lifted using four lugs that are threaded into holes in the top plate of the lid (Holtec Drawing 1495, Section 1.5). It is noted that failure of the lid attachment would not result in any event of safety consequence because a free-falling HI-STORM 100 lid cannot strike a stored MPC (due to its size and orientation). Operational limits on the carry height of the HI-STORM 100 lid above the top of the storage overpack containing a loaded MPC preclude any significant lid rotation out of the horizontal plane in the event of a handling accident. Therefore, contact between the top of the MPC and the edge of a dropped lid due to uncontrolled lowering of the lid during the lid placement operation is judged to be a non-credible scenario. ~~Appendix 3.AC provides an example of a commercially available item that has the appropriate safety factors to serve as a lifting device for the HI-STORM 100 overpack top lid.~~ Except for location of the lift points, the lifting device for the HI-STORM 100S lid is the same as for the regular HI-STORM 100 lid. Since the lid weight for the HI-STORM 100S bounds the HI-STORM 100, the calculated safety factors for the lifting of the HI-STORM 100S lid are reduced and are also reported in the summary table below.

In addition to the HI-STORM 100 top lid lifting analysis, ~~Appendix 3.AC also contains details of the strength qualification of the other lid lifting holes, and associated lid lifting devices, for the HI-TRAC pool lid and top lid has been performed.~~ The qualification is based on the Regulatory Guide 3.61 requirement that a load factor of 3 results in stresses less than the yield stress. ~~Lifting of the HI-TRAC 125 pool lid and top lid are considered in Appendix 3.AC.~~ The results for the HI-TRAC 125 bound the results for the HI-TRAC 125D, and the HI-TRAC 100, since the lid weights used in *the calculation Appendix 3.AC* are greater than or equal to all other HI-TRAC lid weights. In addition, the HI-TRAC 125D has larger diameter lifting holes in its pool lid, which provide greater capacity for lifting. Example commercially available lifting structures are considered ~~in Appendix 3.AC and~~

it is shown that thread engagement lengths are acceptable. Loads to lifting devices are permitted to be at a maximum angle of 45 degrees from vertical. A summary of results from Appendix 3.AC, pertaining to the various lid lifting operations, is given in the table below:

<b>Summary of HI-STORM 100 Lid Lifting Analyses</b>		
<b>Item</b>	<b>Dead Load (lb)</b>	<b>Minimum Safety Factor</b>
HI-STORM 100 (100S) Top Lid Lifting	23,000 (25,500)	1.978 (1.784)
HI-TRAC Pool Lid Lifting	12,500	4.73
HI-TRAC Top Lid Lifting	2,750	11.38

The analysis Appendix 3.AC demonstrates that thread engagement is sufficient for the threaded holes used solely for lid lifting and that commercially available lifting devices engaging the threaded holes, are available. We note that all reported safety factors are based on an allowable strength equal to 33.3% of the yield strength of the lid material when evaluating shear capacity of the internal threads and based on the working loads of the commercially available lifting devices associated with the respective threaded holes.

#### 3.4.3.8 HI-TRAC Pool Lid Analysis - Lifting MPC From the Spent Fuel Pool (Load Case 01 in Table 3.1.5)

During lifting of the MPC from the spent fuel pool, the HI-TRAC pool lid supports the weight of a loaded MPC plus water (see Figure 3.4.21). Appendix 3.AB details the calculations performed to show structural integrity under this condition for both the HI-TRAC 100 and the HI-TRAC 125 transfer casks. In accordance with the general guidelines set down at the beginning of Subsection 3.4.3, the pool lid is considered as both Region A and Region B for evaluating safety factors. The analysis in Appendix 3.AB shows that the stress in the pool lid top plate is less than the Level A allowable stress under pressure equivalent to the heaviest MPC, contained water, and lid self weight (Region B evaluation). Stresses in the lids and bolts are also shown to be below yield under three times the applied lifted load (Region A evaluation using Regulatory Guide 3.61 criteria). The threaded holes in the HI-TRAC pool lid are also examined for acceptable engagement length under the condition of lifting the MPC from the pool. This analysis is performed in Appendix 3.AC. It is demonstrated in Appendix 3.AC that the pool lid peripheral bolts have adequate engagement length into the pool lid to permit the transfer of the required load. The safety factor is defined based on the strength limits imposed by Regulatory Guide 3.61.

The following table summarizes the results of the analyses *for the HI-TRAC pool lid* performed in Appendix 3.AB and the thread engagement calculation in Appendix 3.AC, as well as the results of similar calculations for the HI-TRAC 125D. Results given in the following table compare calculated stress (or load) and allowable stress (or load). In all cases, the safety factor is defined as the allowable value divided by the calculated value.

<b>HI-TRAC Pool Lid Lifting a Loaded MPC Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Bending Stress - HI-TRAC 125 - Region B Analysis - Pool Lid Top Plate	10.1	26.3	2.604
Lid Bending Stress - HI-TRAC 125 - Region B Analysis - Pool Lid Bottom Plate	5.05	26.3	5.208
Lid Bending Stress - HI-TRAC 100 - Region B Analysis- Pool Lid Top Plate	10.06	26.3	2.614
Lid Bending Stress - HI-TRAC 100 - Region B Analysis- Pool Lid Bottom Plate	6.425	26.3	4.093
Lid Bending Stress - HI-TRAC 125D - Region B Analysis - Pool Lid Top Plate	10.1	26.3	2.604
Lid Bending Stress - HI-TRAC 125D - Region B Analysis - Pool Lid Bottom Plate	5.05	26.3	5.208
Lid Bolt Stress - HI-TRAC 125 – (3D*)	18.92	95.0	5.02
Lid Bolt Stress - HI-TRAC 100 – (3D*)	18.21	95.0	5.216
Lid Bolt Force - HI-TRAC 125D – (3D*)	25.77‡	84.05‡	3.262
Lid Bending Stress - HI-TRAC 125 - Region A Analysis - Pool Lid Top Plate (3D*)	30.3	33.15	1.094
Lid Bending Stress - HI-TRAC 125 - Region A Analysis - Pool Lid Bottom Plate (3D*)	15.15	33.15	2.188
Lid Bending Stress –HI-TRAC 100 – Region A Analysis- Pool Lid Top Plate (3D*)	30.19	33.15	1.098
Lid Bending Stress –HI-TRAC 100 – Region A Analysis- Pool Lid Bottom Plate (3D*)	19.28	33.15	1.72
Lid Bending Stress - HI-TRAC 125D - Region A Analysis - Pool Lid Top Plate (3D*)	30.3	33.15	1.094
Lid Bending Stress –HI-TRAC 125D – Region A Analysis- Pool Lid Bottom Plate (3D*)	15.15	33.15	2.188
Lid Thread Engagement Length (HI-TRAC 125)	137.5‡	324.6‡	2.362

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3.

<sup>‡</sup> Calculated and allowable value for this item in (kips).

3.4.3.9 HI-TRAC Transfer Lid Analysis - Lifting MPC Away from Spent Fuel Pool (Load Case 01 in Table 3.1.5)

During transfer to or from a storage overpack using a HI-TRAC 125 or a HI-TRAC 100, the HI-TRAC transfer lid supports the weight of a loaded MPC. Figure 3.4.21 illustrates the lift operation. In accordance with the general lifting analysis guidelines, the transfer lid should be considered as both a Region A (Regulatory Guide 3.61 criteria) and a Region B location (ASME Section III, Subsection NF for Class 3 plate and shell structures) for evaluation of safety factors. ~~Appendices 3.AD and 3.AJ present analyses and results for~~ The HI-TRAC 125 transfer lid and the HI-TRAC 100 transfer lid *are analyzed separately because of differences in geometry, respectively.* The HI-TRAC 125D employs a specially designed mating device in combination with the pool lid to transfer a loaded MPC to or from a storage overpack. Thus, a transfer lid analysis is not performed for the HI-TRAC 125D. Results for the HI-TRAC 125D pool lid are presented in the previous subsection.

~~It is shown in the above-mentioned appendices that the transfer lid doors can support a loaded MPC together with the door weight without exceeding ASME NF stress limits and the more conservative limits of Regulatory Guide 3.61. It is also shown that the connecting structure transfers the load to the cask body without overstress. The following tables summarize the results for both HI-TRAC casks:~~

<b>HI-TRAC 125 Transfer Lid – Lifting Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 125 - Door Plate – (3D*)	9.381	32.7	3.486
HI-TRAC 125 - Door Plate – Region B	3.127	26.25	8.394
HI-TRAC 125 – Wheel Track (3D*)	26.91	36.0	1.338
HI-TRAC 125 - Door Housing Bottom Plate-Region B	7.701	26.25	3.409
HI-TRAC 125 - Door Housing Bottom Plate-(3D*)	23.103	32.7	1.415
HI-TRAC 125 - Door Housing Stiffeners- (3D*)	4.131	32.7	7.913
HI-TRAC 125 - Housing Bolts-Region B	29.96	57.5	1.919
HI-TRAC 125 – Housing Bolts (3D*)	89.88	95.0	1.057
HI-TRAC 125 – Lid Top Plate (3D*)	30.907	32.7	1.058

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3

<b>HI-TRAC 100 Transfer Lid – Lifting Evaluation<sup>†</sup></b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 100 - Door Plate – (3D*)	22.188	32.7	1.474
HI-TRAC 100 - Door Plate – Region B	7.396	26.25	3.549
HI-TRAC 100 – Wheel Track (3D*)	13.011	36.0	2.767
HI-TRAC 100 – Door Housing Bottom Plate- Region B	7.447	26.25	3.525
HI-TRAC 100 – Door Housing Bottom Plate- (3D*)	22.336	32.7	1.464
HI-TRAC 100 – Door Housing Stiffeners- (3D*)	4.917	32.7	6.65
HI-TRAC 100 – Welds Connecting Door Housing Stiffeners (3D*)	11.802	32.7	2.771
HI-TRAC 100 - Housing Bolts-Region B	22.478	57.5	2.558
HI-TRAC 100 – Housing Bolts (3D*)	67.423	95.0	1.409
HI-TRAC 100 – Lid Top Plate (3D*)	19.395	32.7	1.686

<sup>†</sup> Region A and B defined at beginning of Subsection 3.4.3

#### 3.4.3.10 HI-TRAC Bottom Flange Evaluation during Lift (Load Case 01 in Table 3.1.5)

During a lifting operation, the HI-TRAC transfer cask body supports the load of a loaded MPC, and the transfer lid (away from the spent fuel pool) or the pool lid plus contained water (lifting from the spent fuel pool). In either case, the load is transferred to the bottom flange of HI-TRAC through the bolts and a state of stress in the flange and the supporting inner and outer shells is developed. Figure 3.4.21 illustrates the lifting operation. ~~Appendix 3.AE provides the evaluation of~~ This area of the HI-TRAC 125 *is analyzed* to demonstrate that *the* required limits on stress are maintained for both ASME and Regulatory Guide 3.61. The bottom flange is considered as an annular plate subject to a total bolt load acting at the bolt circle and supported by reaction loads developed in the inner and outer shells of HI-TRAC. The solution for maximum flange bending stress is found in the classical literature and stresses and corresponding safety factors developed for the bottom flange and for the outer and inner shell weld shear stress. Since the welds are partial penetration, weld stress evaluation bounds an evaluation of direct stress. The table below summarizes the results of the evaluation ~~in~~ Appendix 3.AE.

<b>Safety Factors in HI-TRAC Bottom Flange During a Lift Operation</b>			
<b>Item</b>	<b>Value(ksi)</b>	<b>Allowable(ksi)</b>	<b>Safety Factor</b>
Bottom Flange – Region B	7.798	26.25	3.37
Bottom Flange (3D*)	23.39	33.15	1.42
Outer Shell (3D*)	4.773	33.15	6.94



The results above bound the results for the HI-TRAC 125D since the dimensions used as input in ~~Appendix 3.AE~~ for the inner shell, the outer shell, and the bottom flange (including the bolt circle diameter) are the same in both the HI-TRAC 125 and 125D transfer casks. In addition, the bottom flange of the HI-TRAC 125D is reinforced by eight gusset plates, whereas the HI-TRAC 125 bottom flange is not reinforced.

#### 3.4.3.11 Conclusion

Synopses of lifting device, device/component interface, and component stresses, under all contemplated lifting operations for the HI-STORM 100 System have been presented in the foregoing. The HI-STORM storage overpack and the HI-TRAC transfer cask have been evaluated for limiting stress states. The results show that all factors of safety are greater than 1.

#### 3.4.4 Heat

The thermal evaluation of the HI-STORM 100 System is reported in Chapter 4.

##### 3.4.4.1 Summary of Pressures and Temperatures

Design pressures and design temperatures for all conditions of storage are listed in Tables 2.2.1 and 2.2.3, respectively.

##### 3.4.4.2 Differential Thermal Expansion

Consistent with the requirements of Reg. Guide 3.61, Load Cases F1 (Table 3.1.3) and E4 (Table 3.1.4) are defined to study the effect of differential thermal expansion among the constituent components in the HI-STORM 100 System. ~~Tables 4.4.9, 4.4.10, 4.4.26, 4.4.27, and 4.4.36 provide~~ the temperatures necessary to perform the differential thermal expansion analyses for the MPC in the HI-STORM 100 and HI-TRAC casks ~~are provided in Chapter 4, respectively.~~ The material presented in ~~the remainder of this paragraph~~ *Subsection 4.4.5* demonstrates that a physical interference between discrete components of the HI-STORM 100 System (e.g., storage overpack and enclosure vessel) will not develop due to differential thermal expansion during any operating condition.

###### 3.4.4.2.1 Normal Hot Environment

Closed form calculations are performed *in Subsection 4.4.5* to demonstrate that initial gaps between the HI-STORM 100 storage overpack or the HI-TRAC transfer cask and the MPC canister, and between the MPC canister and the fuel basket, will not close due to thermal expansion of the system components under loading conditions, defined as F1 and E4 in Tables 3.1.3 and 3.1.4, respectively. To assess this in the most conservative manner, the thermal solutions computed in Chapter 4, including the thermosiphon effect, are surveyed for the following information.

- The radial temperature distribution in each of the fuel baskets at the location of peak center metal temperature.

- The highest and lowest mean temperatures of the canister shell for the hot environment condition.
- ~~The inner and outer surface temperature of the HI-STORM 100 storage overpack and the HI-TRAC transfer cask at the location of highest and lowest surface temperature (which will produce the lowest mean temperature).~~

Tables 4.4.9, 4.4.10, ~~and 4.4.26, and 4.4.27, 4.4.27, and 4.4.36~~ present the resulting temperatures used in the evaluation of the MPC expansion in the HI-STORM 100 storage overpack. Table 4.5.2 presents similar results for the MPC in the HI-TRAC transfer cask.

Using the temperature information in the above-mentioned tables, simplified thermoelastic solutions of equivalent axisymmetric problems are used to obtain conservative estimates of gap closures. The following procedure, which conservatively neglects axial variations in temperature distribution, is utilized.

1. Use the surface temperature information for the fuel basket to define a parabolic distribution in the fuel basket that bounds (from above) the actual temperature distribution. Using this result, generate a conservatively high estimate of the radial and axial growth of the different fuel baskets using classical closed form solutions for thermoelastic deformation in cylindrical bodies.
2. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-basket gaps.
3. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-storage overpack and canister-to-HI-TRAC gaps.
4. ~~Use the storage overpack and HI-TRAC surface temperatures to construct a logarithmic temperature distribution (characteristic of a thick walled cylinder) at the location used for canister thermal growth calculations; and use this distribution to predict an estimate of storage overpack or HI-TRAC (as applicable) radial and axial growth.~~
45. For given initial clearances, compute the operating clearances.

~~The calculation procedure outlined above is used in Appendix 3.I (HI-TRAC), and in Appendices 3.U, 3.V, 3.W, and 3.AQ (HI-STORM 100 storage overpack with MPC 24, MPC 32, MPC 68, and 24E respectively). The results are summarized in the tables given below Subsection 4.4.5 for normal storage conditions. It can be verified by referring to the Design Drawings provided in Section 1.5 of this report and Subsection 4.4.5, that the clearances between the MPC basket and canister structure, as well as that between the MPC shell and storage overpack or HI-TRAC inside surface, are sufficient to preclude a temperature induced interference from differential thermal expansions under normal operating conditions.~~

The worst case MPC is evaluated in the HI-TRAC transfer cask, in lieu of all MPC designs. In all cases, the minimal initial radial gap between MPC and overpack is used as the initial point.

THERMOELASTIC DISPLACEMENTS IN THE MPC AND HI-STORM 100 STORAGE OVERPACK UNDER HOT TEMPERATURE ENVIRONMENT CONDITION				
CANISTER – FUEL BASKET				
	Radial Direction (in.)		Axial Direction (in.)	
Unit	Initial Clearance	Final Clearance	Initial Clearance	Final Clearance
MPC-24	0.1875	0.1048	1.8125	1.404
MPC-24E	0.1875	0.104	1.8125	1.404
MPC-32	0.1875	0.103	1.8125	1.398
MPC-68	0.1875	0.091	1.8125	1.336
CANISTER – STORAGE OVERPACK				
	Radial Direction (in.)		Axial Direction (in.)	
Unit	Initial Clearance	Final Clearance	Initial Clearance	Final Clearance
MPC-24	0.5	0.435	1.0	0.633
MPC-24E	0.5	0.434	1.0	0.628
MPC-32	0.5	0.433	1.0	0.621
MPC-68	0.5	0.434	1.0	0.628
THERMOELASTIC DISPLACEMENTS IN THE MPC AND HI-TRAC UNDER HOT TEMPERATURE ENVIRONMENT CONDITION				
CANISTER – FUEL BASKET				
	Radial Direction (in.)		Axial Direction (in.)	
Unit	Initial Clearance	Final Clearance	Initial Clearance	Final Clearance
MPC (worst case)	0.1875	0.083	1.8125	1.305
CANISTER – HI-TRAC				
	Radial Direction (in.)		Axial Direction (in.)	
Unit	Initial Clearance	Final Clearance	Initial Clearance	Final Clearance
MPC (worst case)	0.125	0.123	0.75	0.735

It can be verified by referring to the Design Drawings provided in Section 1.5 of this report and the foregoing table, that the clearances between the MPC basket and canister structure, as well as that between the MPC shell and storage overpack or HI-TRAC inside surface, are sufficient to preclude a temperature induced interference from differential thermal expansions under normal operating conditions.

### 3.4.4.2.2 Fire Accident

It is shown in Chapter 11 that the fire accident has a small effect on the MPC temperatures because of the short duration of the fire accidents and the large thermal inertia of the storage overpack. Therefore, a structural evaluation of the MPC under the postulated fire event is not required. The conclusions reached in Subsection 3.4.4.2.1 are also appropriate for the fire accident with the MPC housed in the storage overpack. Analysis of fire accident temperatures of the MPC housed within the HI-TRAC for thermal expansion is unnecessary, as the HI-TRAC, directly exposed to the fire, expands to increase the gap between the HI-TRAC and MPC.

As expected, the external surfaces of the HI-STORM 100 storage overpack that are directly exposed to the fire event experience maximum rise in temperature. The outer shell and top plate in the top lid are the external surfaces that are in direct contact with heated air from fire. The table below, extracted from data provided in Chapter 11, provides the maximum temperatures attained at the key locations in HI-STORM 100 storage overpack under the postulated fire event.

<b>Component</b>	<b>Maximum Fire Condition Temperature (Deg. F)</b>
Storage Overpack Inner Shell	300
Storage Overpack Radial Concrete Mid-Depth	<del>184235173</del>
Storage Overpack Outer Shell	<del>570585570</del>
Storage Overpack Lid	<del>&lt;570585570</del>

The following conclusions are readily reached from the above table.

- The maximum metal temperature of the carbon steel shell most directly exposed to the combustion air is well below 600°F (Table 2.2.3 applicable short-term temperature limit). 600°F is well below the permissible temperature limit in the ASME Code for the outer shell material.
- The bulk temperature of concrete is well below the normal condition temperature limit of 300°F specified in Table 2.2.3 and Appendix 1.D. ACI-349 permits 350°F as the short-term temperature limit; the shielding concrete in the HI-STORM 100 Overpack, as noted in Appendix 1.D, will comply with the specified compositional and manufacturing provisions of ACI-349. As the detailed information in Section 11.2 shows, the radial extent in the concrete where the local temperature exceeds 350°F begins at the outer shell/concrete interface and ends in less than one-inch. Therefore, the potential loss in the shielding material's effectiveness is less than 4% of the concrete shielding mass in the overpack annulus.
- The metal temperature of the inner shell does not exceed 300°F at any location, which is below the ~~accident normal~~ condition temperature limit of ~~400350~~ 350°F specified in Table 2.2.3 for the inner shell.

- The presence of a stitch weld between the overpack inner shell and the overpack top plate ensures that there will be no pressure buildup in the concrete annulus due to the concrete losing water that then turns to steam.

The above summary confirms that the postulated fire event will not jeopardize the structural integrity of the HI-STORM 100 Overpack or significantly diminish its shielding effectiveness.

The above conclusions, as relevant, also apply to the HI-TRAC fire considered in Chapter 11. Water jacket over-pressurization is precluded by the safety valve set point. The non-structural effects of loss of water have been evaluated in Chapter 5 and shown to meet regulatory limits. Therefore, it is concluded that the postulated fire event will not cause significant loss in storage overpack or HI-TRAC shielding function.

#### 3.4.4.3 Stress Calculations

This subsection presents calculations of the stresses in the different components of the HI-STORM 100 System from the effects of mechanical load case assembled in Section 3.1. Loading cases for the MPC fuel basket, the MPC enclosure vessel, the HI-STORM 100 storage overpack and the HI-TRAC transfer cask are listed in Tables 3.1.3 through 3.1.5, respectively. The load case identifiers defined in Tables 3.1.3 through 3.1.5 denote the cases considered.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability of fuel from the MPC and the MPC from the HI-STORM 100 storage overpack or from the HI-TRAC transfer cask.

For all stress evaluations, the allowable stresses and stress intensities for the various HI-STORM 100 System components are based on bounding high metal temperatures to provide additional conservatism (Table 3.1.17 for the MPC basket, for example).

In addition to the loading cases germane to stress evaluations mentioned above, three cases pertaining to the stability of HI-STORM 100 are also considered (Table 3.1.1).

The results of various stress calculations on components are reported. The calculations are either performed directly as part of the text, or *carried out in a separate calculation report* ~~are summarized in an appendix (see the list of all supporting appendices provided in Section 3.6)~~ that provides details of strength of materials evaluations or finite element numerical analysis. The specific calculations reported in this subsection are:

1. MPC stress calculations
2. HI-STORM 100 storage overpack stress calculations
3. HI-TRAC stress calculations

The MPC calculations reported in this document are complemented by analyses in the HI-STAR 100 Dockets. As noted earlier in this chapter, calculations for MPC components that are reported in HI-STAR 100 FSAR and SAR (Docket Numbers 72-1008 or 71-9261) are not repeated here unless geometry or load changes warrant reanalysis. For example, analysis of the MPC lid is not included in this submittal since neither the MPC lid loading nor geometry is affected by the MPC being placed in HI-TRAC or HI-STORM 100. MPC stress analyses reported herein focus on the basket and canister stress distributions due to the design basis (45g) lateral deceleration imposed by a non-mechanistic tip-over of the HI-STORM 100 storage overpack or a horizontal drop of HI-TRAC. In the submittals for the HI-STAR 100 FSAR and SAR (Docket Numbers 72-1008 and 71-9261, for storage and transport, respectively), the design basis deceleration was 60g. In this submittal the design basis deceleration is 45g. However, since the geometry of the MPC external boundary condition, viz. canister-to-storage overpack gap, has changed, a reanalysis of the MPC stresses under the lateral deceleration loads is required. This analysis is performed and the results are summarized in this subsection.

The HI-STORM 100 storage overpack and the HI-TRAC transfer cask have been evaluated for certain limiting load conditions that are germane to the storage and operational modes specified for the system in Tables 3.1.1 and 3.1.5. The determination of component safety factors at the locations considered in the HI-STORM 100 storage overpack and in the HI-TRAC transfer cask is based on the allowable stresses permitted by the ASME Code Section III, Subsection NF for Class 3 plate and shell support structures.

#### 3.4.4.3.1 MPC Stress Calculations

The structural function of the MPC in the storage mode is stated in Section 3.1. The calculations presented here demonstrate the ability of the MPC to perform its structural function. The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, or impairment of ready retrievability.

##### 3.4.4.3.1.1 Analysis of Load Cases E.3.b, E.3.c (Table 3.1.4) and F.3.b, F.3.c (Table 3.1.3)

Analyses are performed for each of the MPC designs. The following subsections describe the model, individual loads, load combinations, and analysis procedures applicable to the MPC. Unfortunately, unlike vertical loading cases, where the analyses performed in the HI-STAR 100 dockets remain fully applicable for application in HI-STORM 100, the response of the MPC to a horizontal loading event is storage overpack-geometry dependent. Under a horizontal drop event, for example, the MPC and the fuel basket structure will tend to flatten. The restraint to this flattening offered by the storage overpack will clearly depend on the difference in the diameters of the storage overpack internal cavity and that of the outer surface of the MPC. In the HI-STORM 100 storage overpack, the diameter difference is larger than that in HI-STAR 100; therefore, the external restraint to MPC ovalization under a horizontal drop event is less effective. For this reason, the MPC stress analysis for lateral loading scenarios must be performed anew for the HI-STORM 100 storage overpack; the results from the HI-STAR 100 analyses will not be conservative. The HI-TRAC transfer casks and HI-STAR 100 overpack inner diameters are identical. Therefore, the analysis of the MPC in the HI-STAR 100 overpack under 60g's for the side impact (Docket 72-1008) bounds the analysis of the MPC in the HI-TRAC under 45g's.

## Description of Finite Element Models of the MPCs Under Lateral Loading

A finite element model of each MPC is used to assess the effects of the accident loads. The models are constructed using ANSYS [3.4.1], and they are identical to the models used in Holtec's HI-STAR 100 submittals in Docket Numbers 72-1008 and 71-9261. The following model description is common to all MPCs.

The MPC structural model is two-dimensional. It represents a one-inch long cross section of the MPC fuel basket and MPC canister.

The MPC model includes the fuel basket, the basket support structures, and the MPC shell. A basket support is defined as any structural member that is welded to the inside surface of the MPC shell. A portion of the storage overpack inner surface is modeled to provide the correct restraint conditions for the MPC. Figures 3.4.1 through 3.4.9 show typical MPC models. The fuel basket support structure shown in the figures is a multi-plate structure consisting of solid shims or support members having two separate compressive load supporting members. For conservatism in the finite element model some dual path compression members (i.e., "V" angles) are simulated as single columns. Therefore, the calculated stress intensities in the fuel basket angle supports from the finite element solution are conservatively overestimated in some locations.

The ANSYS model is not intended to resolve the detailed stress distributions in weld areas. Individual welds are not included in the finite element model. A separate analysis for basket welds and for the basket support "V" angles is *performed outside of ANSYS* ~~contained in Appendix 3.Y.~~

No credit is taken for any load support offered by the *neutron absorber Boral* panels, sheathing, and the aluminum heat conduction elements. Therefore, these so-called non-structural members are not represented in the model. The bounding MPC weight used, however, does include the mass contributions of these non-structural components.

The model is built using five ANSYS element types: BEAM3, PLANE82, CONTAC12, CONTAC26, and COMBIN14. The fuel basket and MPC shell are modeled entirely with two-dimensional beam elements (BEAM3). Plate-type basket supports are also modeled with BEAM3 elements. Eight-node plane elements (PLANE82) are used for the solid-type basket supports. The gaps between the fuel basket and the basket supports are represented by two-dimensional point-to-point contact elements (CONTAC12). Contact between the MPC shell and the storage overpack is modeled using two-dimensional point-to-ground contact elements (CONTAC26) with an appropriate clearance gap.

Two orientations of the deceleration vector are considered. The 0-degree drop model includes the storage overpack-MPC interface in the basket orientation illustrated in Figure 3.1.2. The 45-degree drop model represents the storage overpack-MPC interface with the basket oriented in the manner of Figure 3.1.3. The 0-degree and the 45-degree drop models are shown in Figures 3.4.1 through 3.4.6. Table 3.4.1 lists the element types and number of elements for current MPC's.

A contact surface is provided in the model ~~is~~ used for drop analyses to represent *the interface between* the storage overpack channels *and the MPC*. As the MPC makes contact with the storage overpack, the MPC shell deforms to mate with the channels that are welded at equal intervals around the storage overpack inner surface. The nodes that define the elements representing the fuel basket and the MPC shell are located along the centerline of the plate material. As a result, the line of nodes that forms the perimeter of the MPC shell is inset from the real boundary by a distance that is equal to half of the shell thickness. In order to maintain the specified MPC shell/storage overpack gap dimension, the radius of the storage overpack channels is decreased by an equal amount in the model.

The three discrete components of the HI-STORM 100 System, namely the fuel basket, the MPC shell, and the storage overpack or HI-TRAC transfer cask, are engineered with small diametral clearances which are large enough to permit unconstrained thermal expansion of the three components under the rated (maximum) heat duty condition. A small diametral gap under ambient conditions is also necessary to assemble the system without physical interference between the contiguous surfaces of the three components. The required gap to ensure unrestricted thermal expansion between the basket and the MPC shell is small and will further decrease under maximum heat load conditions, but will introduce a physical nonlinearity in the structural events involving lateral loading (such as side drop of the system) under ambient conditions. It is evident from the system design drawings that the fuel basket that is non-radially symmetric is in proximate contact with the MPC shell at a discrete number of locations along the circumferences. At these locations, the MPC shell, backed by the channels attached to the storage overpack, provides a support line to the fuel basket during lateral drop events. Because the fuel basket, the MPC shell, and the storage overpack or HI-TRAC are all three-dimensional structural weldments, their inter-body clearances may be somewhat uneven at different azimuthal locations. As the lateral loading is increased, clearances close at the support locations, resulting in the activation of the support from the storage overpack or HI-TRAC.

The bending stresses in the basket and the MPC shell at low lateral loading levels which are too small to close the support location clearances are secondary stresses since further increase in the loading will activate the storage overpack's or HI-TRAC's transfer cask support action, mitigating further increase in the stress. Therefore, to compute primary stresses in the basket and the MPC shell under lateral drop events, the gaps should be assumed to be closed. However, in the analyses, we have conservatively assumed that an initial gap of 0.1875" exists, in the direction of the applied deceleration, at all support locations between the fuel basket and the MPC shell and that the clearance gap between the shell and the storage overpack at the support locations is 3/16". In the evaluation of safety factors for the MPC-24, MPC-32, and MPC-68, the total stress state produced by the applied loading on these configurations is conservatively compared with primary stress levels, even though the self-limiting stresses should be considered secondary in the strict definition of the Code. To illustrate the conservatism,- we have eliminated the secondary stress (that develops to close the clearances) in the comparison with primary stress allowable values and report safety factors for the MPC-24E that are based only on primary stresses necessary to maintain equilibrium with the inertia forces.



ANSYS requires that for a static solution all bodies be constrained to prevent rigid body motion. Therefore, in the 0 degree and 45 degree drop models, two-dimensional linear spring elements (COMBIN14) join the various model components, i.e., fuel basket and enclosure vessel, at the point of initial contact. This provides the necessary constraints for the model components in the direction of the impact. By locating the springs at the points of initial contact, where the gaps remain closed, the behavior of the springs is identical to the behavior of a contact element. Linear springs and contact elements that connect the same two components have equal stiffness values.

### Description of Individual Loads and Boundary Conditions Applied to the MPCs

The method of applying each individual load to the MPC model is described in this subsection. The individual loads are listed in Table 2.2.14. A free-body diagram of the MPC corresponding to each individual load is given in Figures 3.4.7-3.4.9. In the following discussion, reference to vertical and horizontal orientations is made. Vertical refers to the direction along the cask axis, and horizontal refers to a radial direction.

Quasi-static structural analysis methods are used. The effects of any dynamic load factors (DLFs) are included in the final evaluation of safety factors. All analyses are carried out using the design basis decelerations in Table 3.1.2.

The MPC models used for side drop evaluations are shown in Figures 3.4.1 through 3.4.6. In each model, the fuel basket and the enclosure vessel are constrained to move only in the direction that is parallel to the acceleration vector. The storage overpack inner shell, which is defined by three nodes needed to represent the contact surface, is fixed in all degrees of freedom. The fuel basket, enclosure vessel, and storage overpack inner shell -are all connected at one location by linear springs, as described in Subsection 3.4.4.3.1.1 (see Figure 3.4.1, for example). Detailed side drop evaluations here focus on an MPC within a HI-STORM 100 storage overpack. Since the analyses performed in Docket Number 72-1008 for the side drop condition in the HI-STAR 100 storage overpack demonstrates a safe condition under a 60g deceleration, no new analysis is required for the MPC and contained fuel basket and fuel during a side drop in the HI-TRAC, which is limited to a 45g deceleration (HI-TRAC and HI-STAR 100 overpacks have the same inside dimensions).

### Accelerations

During a side impact event, the stored fuel is directly supported by the cell walls in the fuel basket. Depending on the orientation of the drop, 0 or 45 degrees (see Figures 3.4.8 and 3.4.9), the fuel is supported by either one or two walls. In the finite element model this load is effected by applying a uniformly distributed pressure over the full span of the supporting walls. The magnitude of the pressure is determined by the weight of the fuel assembly (Table 2.1.6), the axial length of the fuel basket support structure, the width of the cell wall, and the impact acceleration. It is assumed that the load is evenly distributed along an axial length of basket equal to the fuel basket support structure. For example, the pressure applied to an impacted cell wall during a 0-degree side drop event is calculated as follows:

$$p = \frac{a_n W}{L c}$$

where:

p = pressure

$a_n$  = ratio of the impact acceleration to the gravitational acceleration

W = weight of a stored fuel assembly

L = axial length of the fuel basket support structure

c = width of a cell wall

For the case of a 45-degree side drop the pressure on any cell wall equals p (defined above) divided by the square root of 2.

It is evident from the above that the effect of deceleration on the fuel basket and canister metal structure is accounted for by amplifying the gravity field in the appropriate direction.

#### Internal Pressure

Design internal pressure is applied to the MPC model. The inside surface of the enclosure vessel shell is loaded with pressure. The magnitude of the internal pressure applied to the model is taken from Table 2.2.1.

For this load condition, the center node of the fuel basket is fixed in all degrees of freedom to numerically satisfy equilibrium.

#### Temperature

Temperature distributions are developed in Chapter 4 and applied as nodal temperatures to the finite element model of the MPC enclosure vessel (confinement boundary). Maximum design heat load has been used to develop the temperature distribution used to demonstrate compliance with ASME Code stress intensity levels.

#### Analysis Procedure

The analysis procedure for this set of load cases is as follows:

1. The stress intensity and deformation field due to the combined loads is determined by the finite element solution. Results are postprocessed and tabulated in the calculation package associated with this FSAR.
2. The results for each load combination are compared to allowables. The comparison with allowable values is made in Subsection 3.4.4.4.

### 3.4.4.3.1.2 Analysis of Load Cases E1.a and E1.c (Table 3.1.4)

Since the MPC shell is a pressure vessel, the classical Lamé's calculations should be performed to demonstrate the shell's performance as a pressure vessel. We note that dead load has an insignificant effect on this stress state. We first perform calculations for the shell under internal pressure. Subsequently, we examine the entire confinement boundary as a pressure vessel subject to both internal pressure and temperature gradients. Finally, we perform confirmatory hand calculations to gain confidence in the finite element predictions.

The stress from internal pressure is found for normal and accident pressures conditions using classical formulas:

Define the following quantities:

$P$  = pressure,  $r$  = MPC radius, and  $t$  = shell thickness.

Using classical thin shell theory, the circumferential stress,  $\sigma_1 = Pr/t$ , the axial stress  $\sigma_2 = Pr/2t$ , and the radial stress  $\sigma_3 = -P$  are computed for both normal and accident internal pressures. The results are given in the following table (conservatively using the outer radius for  $r$ ):

<b>Classical Shell Theory Results for Normal and Accident Internal Pressures</b>				
<b>Item</b>	<b><math>\sigma_1</math> (psi)</b>	<b><math>\sigma_2</math> (psi)</b>	<b><math>\sigma_3</math> (psi)</b>	<b><math>\sigma_1 - \sigma_3</math> (psi)</b>
P= 100 psi	6838	3419	-100	6938
P= 200 psi	13675	6838	-200	13875

#### Finite Element Analysis (Load Case E1.a and E1.c of Table 3.1.4)

The MPC shell, the top lid, and the baseplate together form the confinement boundary (enclosure vessel) for storage of spent nuclear fuel. In this section, we evaluate the operating condition consisting of dead weight, internal pressure, and thermal effects for the hot condition of storage. The top and bottom plates of the MPC enclosure vessel (EV) are modeled using plane axisymmetric elements, while the shell is modeled using the axisymmetric thin shell element. The thickness of the top lid varies in the different MPC types; for conservative results, the minimum thickness top lid is modeled. The temperature distributions for all MPC constructions are nearly identical in magnitude and gradient and reflect the thermosiphon effect inside the MPC. Temperature differences across the thickness of both the baseplate and the top lid exist during HI-STORM 100's operations. There is also a thermal gradient from the center of the top lid and baseplate out to the shell wall. The metal temperature profile is essentially parabolic from the centerline of the MPC out to the MPC shell. There is also a parabolic temperature profile along the length of the MPC canister. Figure 3.4.11 shows a sketch of the confinement boundary structure with identifiers A-I locating points where temperature input data is used to represent a continuous temperature distribution for analysis purposes. The overall dimensions of the confinement boundary are also shown in the figure.

The ~~desired~~ temperatures for confinement thermal stress analysis are determined from *the thermal numerical analyses that support Tables 4.4.9, 4.4.10, 4.4.19, and 4.4.26, and 4.4.27, and 4.4.27* in Chapter 4. The MPC-68 is identified to have the maximum through thickness thermal gradients. ~~Detailed stress analyses are performed only for the MPC-68; these results are representative for all MPCs.~~ For conservatism, a bounding temperature profile is defined for all MPC types and used as input for thermal stress analysis.

Figure 3.4.12 shows details of the finite element model of the top lid, canister shell, and baseplate. The top lid is modeled with 40 axisymmetric quadrilateral elements; the weld connecting the lid to the shell is modeled by a single element solely to capture the effect of the top lid attachment to the canister offset from the middle surface of the top lid. The MPC canister is modeled by 50 axisymmetric shell elements, with 20 elements concentrated in a short length of shell appropriate to capture the so-called "bending boundary layer" at both the top and bottom ends of the canister. The remaining 10 shell elements model the MPC canister structure away from the shell ends in the region where stress gradients are expected to be of less importance. The baseplate is modeled by 20 axisymmetric quadrilateral elements. Deformation compatibility at the connections is enforced at the top by the single weld element, and deformation and rotation compatibility at the bottom by additional shell elements between nodes 106-107 and 107-108.

The geometry of the model is listed below (terms are defined in Figure 3.4.12):

$H_t =$	9.5" (the minimum thickness lid is assumed)
$R_L =$	0.5 x 67.25" (Bill of Materials for Top Lid)
$L_{MPC} =$	190.5" (Drawing 1996, Sheet 1)
$t_s =$	0.5"
$t_{BP} =$	0.5 x 68.375"
$\beta =$	$2\sqrt{R_s t_s} \approx 12"$ (the "bending boundary layer")

Stress analysis results are obtained for two cases as follows:

- a. internal pressure = 100 psi
- b. internal pressure = 100 psi plus applied temperatures

For this configuration, dead weight of the top lid acts to reduce the stresses due to pressure. For example, the equivalent pressure simulating the effect of the weight of the top lid is an external pressure of 3 psi, which reduces the pressure difference across the top lid to 97 psi. The dead weight of the top lid is neglected to provide additional conservatism in the results. The dead weight of the baseplate, however, adds approximately 0.73 psi to the effective internal pressure acting on the base.

The effect of dead weight is still insignificant compared to the 100 psi design pressure, and is therefore neglected. The thermal loading in the confinement vessel is obtained by developing a parabolic temperature profile to the entire length of the MPC canister and to the top lid and baseplate. The temperature data provided at locations A-I in Figure 3.4.11 and 3.4.12 are sufficient to establish the profiles. Through-thickness temperatures are assumed linearly interpolated between top and bottom surfaces of the top lid and baseplate.

Finally, in the analysis, all material properties and expansion coefficients are considered to be temperature-dependent in the model.

Results for stress intensity are reported for the case of internal pressure alone and for the combined loading of pressure plus temperature (Load Case E1.c in Table 3.1.4). Tables 3.4.7 and 3.4.8 report results at the inside and outside surfaces of the top lid and baseplate at the centerline and at the extreme radius. Canister results are reported in the "bending boundary layer" and at a location near mid-length of the MPC canister. In the tables, the calculated value is the value from the finite element analysis, the categories are  $P_m$  = primary membrane;  $P_L + P_b$  = local membrane plus primary bending; and  $P_L + P_b + Q$  = primary plus secondary stress intensity. The allowable strength value is obtained from the appropriate table in Section 3.1 for Level A conditions, and the safety factor SF is defined as the allowable strength divided by the calculated value. Allowable strengths for Alloy X are taken at ~~550300~~ 400 degrees F, 400 degrees F, and 500 degrees F, respectively, for the MPC lid, baseplate, and canister shell ~~at the bottom of the MPC and 500 degrees F at the top of the MPC.~~ These temperatures reflect actual operating conditions per Table 4.4.19. The results given in Tables 3.4.7 and 3.4.8 demonstrate the ruggedness of the MPC as a confinement boundary.

The results in Table 3.4.7 and 3.4.8 also show that the baseplate and the shell connection to the baseplate are the most highly stressed regions under the action of internal pressure. To confirm the finite element results, we perform an alternate closed form solution using classical plate and shell theory equations that are listed in or developed from the reference (Timoshenko and Woinowsky-Krieger, Theory of Plate and Shells, McGraw Hill, Third Edition).

Assuming that the thick baseplate receives little support against rotation from the thin shell, the bending stress at the centerline is evaluated by considering a simply supported plate of radius  $a$  and thickness  $h$ , subjected to lateral pressure  $p$ . The maximum bending stress is given by

$$\sigma = \frac{3(3 + \nu)}{8} p \left(\frac{a}{h}\right)^2$$

where:

$$a = .5 \times 68.375''$$

$$h = 2.5''$$

$$\nu = 0.3 \text{ (Poisson's Ratio)}$$

$$p = 100 \text{ psi}$$

Calculating the stress in the plate gives  $\sigma = 23,142$  psi.

Now consider the thin MPC shell ( $t = 0.5$ " ) and first assume that the baseplate provides a clamped support to the shell. Under this condition, the bending stress in the thin shell at the connection to the plate is given as

$$\sigma_{Bp} = 3 p \frac{a}{t} \frac{(1-\nu/2)}{\sqrt{3} (1-\nu^2)^{1/2}} = 10,553 \text{ psi}$$

In addition to this stress, there is a component of stress in the shell due to the baseplate rotation that causes the shell to rotate. The joint rotation is essentially driven by the behavior of the baseplate as a simply supported plate; the shell offers little resistance because of the disparity in thickness and will essentially follow the rotation of the thick plate.

Using formulas from thin shell theory, the additional axial bending stress in the shell due to this rotation  $\theta$  can be written in the form

$$\sigma_{B\theta} = 12 \beta D_s \frac{\theta}{t^2}$$

where

$$\theta = pa^3/8D(1+\nu) * \left( \frac{1}{1+\alpha} \right)$$

and

$$D = \frac{Eh^3}{12(1-\nu^2)} \quad E = \text{plate Young's Modulus}$$

$$\alpha = \frac{2\beta at^3}{h^3(1+\nu)}$$

$$D_s = \frac{Et^3}{12(1-\nu^2)}$$

$$\beta^2 = \sqrt{3(1 - \nu^2)} / at$$

Substituting the numerical values gives

$$\sigma_{B0} = 40,563 \text{ psi}$$

We note that the approximate solution is independent of the value chosen for Young's Modulus as long as the material properties for the plate and shell are the same.

Combining the two contributions to the shell bending stress gives the total extreme fiber stress in the longitudinal direction as 51,116 psi.

The baseplate stress value, 23,142 psi, compares well with the finite element result 20,601 ~~20,528~~ psi (Table 3.4.7). The shell joint stress, 51,116 psi, is greater than the finite element result (43,645 ~~43,986~~ psi in Table 3.4.7). This is due to the local effects of the shell-to-baseplate connection offset. That is, the connection between shell and baseplate in the finite element model is at the surface of the baseplate, not at the middle surface of the baseplate. This offset will cause an additional bending moment that will reduce the rotation of the plate and hence, reduce the stress in the shell due to the rotation of the baseplate.

In summary, the approximate closed form solution confirms the accuracy of the finite element analysis in the baseplate region.

*From Table 2.2.1, the off-normal design internal pressure is 110 psi, or ten percent greater than the normal design pressure. Whereas Level A service limits are used to establish allowables for the normal design pressure, Level B service limits are used for off-normal loads. Since Subsection NB of the ASME Code permits an identical 10% increase in allowable stress intensity values for primary stress intensities generated by Level B Service Loadings, it stands to reason that the safety factors reported in Tables 3.4.7 and 3.4.8 bound the case of off-normal design internal pressure.*

Under the accident pressure, the MPC baseplate experiences bending. Table NB-3217-1 permits the bending stress at the outer periphery of the baseplate and in the shell wall at the connection to be considered as a secondary bending stress if the primary bending stress at the center of the baseplate can be shown to meet the stress limits without recourse to the restraint provided by the MPC shell. To this end, the bending stress at the center of the baseplate is computed in a conservative manner assuming the baseplate is simply supported at the periphery. The bending stress for a simply supported circular plate is

$$\sigma = (9/8)p \left( \frac{r}{t} \right)^2$$

At the accident pressure, conservatively set at twice the normal operating pressure, the maximum stress is:

Bending stress at center of baseplate = 46,284 psi

Since this occurrence is treated as a Level D event, the stress intensity is compared with the limit from Table 3.1.14 and the safety factor computed as, “SF”, where

$$SF = 67,49,300 \text{ psi}/(46,284+200) \text{ psi} = 1.459$$

#### 3.4.4.3.1.3 Elastic Stability and Yielding of the MPC Basket under Compression Loads (Load Case F3 in Table 3.1.3)

This load case corresponds to the scenario wherein the loaded MPC is postulated to drop causing a compression state in the fuel basket panels.

##### a. Elastic Stability

Following the provisions of Appendix F of the ASME Code [3.4.3] for stability analysis of Subsection NG structures, (F-1331.5(a)(1)), a comprehensive buckling analysis is performed using ANSYS. For this analysis, ANSYS's large deformation capabilities are used. This feature allows ANSYS to account for large nodal rotations in the fuel basket, which are characteristic of column buckling. The interaction between compressive and lateral loading, caused by the deformation, is exactly included. Subsequent to the large deformation analysis, the basket panel that is most susceptible to buckling failure is identified by a review of the results. The lateral displacement of a node located at the mid-span of the panel is measured for the range of impact decelerations. The buckling or collapse load is defined as the impact deceleration for which a slight increase in its magnitude results in a disproportionate increase in the lateral displacement.

The stability requirement for the MPC fuel basket under lateral loading is satisfied if two-thirds of the collapse deceleration load is greater than the design basis horizontal acceleration (Table 3.1.2). This analysis was performed for the HI-STAR 100 submittal (Docket Number 72-1008) under a 60g deceleration loading. Within the HI-STAR 100 FSAR (Docket Number 72-1008), Figures 3.4.27 through 3.4.32 are plots of lateral displacement versus impact deceleration for the MPC-24, MPC-32, and MPC-68. It should be noted that the displacements (in the HI-STAR 100 FSAR) in Figures 3.4.27 through 3.4.31 are expressed in  $1 \times 10^{-1}$  inch and Figure 3.4.32 is expressed in  $1 \times 10^{-2}$  inch. The plots in the HI-STAR 100 FSAR clearly show that the large deflection collapse load of the MPC fuel basket is greater than 1.5 times the design basis deceleration for all baskets in all orientations. The results for the MPC-24E are similar. Thus, the requirements of Appendix F are met for lateral deceleration loading under Subsection NG stress limits for faulted conditions.

An alternative solution for the stability of the fuel basket panel is obtained using the methodology espoused in NUREG/CR-6322 [3.4.13]. In particular, we consider the fuel basket panels as wide plates in accordance with Section 5 of NUREG/CR-6322. We use eq.(19) in that section with the “K” factor set to the value appropriate to a clamped panel. Material properties are selected corresponding to a metal temperature of 500 degrees F which bounds computed metal temperatures



at the periphery of the basket. In general, the basket periphery sees the largest loading in an impact scenario. The critical buckling stress is:

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

where h is the panel thickness, a is the unsupported panel length, E is the Young's Modulus of Alloy X at 500 degrees F, v is Poisson's Ratio, and K=0.65 (per Figure 6 of NUREG/CR-6322).

The MPC-24 has a small h/a ratio; the results of the finite element stress analyses under design basis deceleration load show that this basket is subject to the highest compressive load in the panel. Therefore, the critical buckling load is computed using the geometry of the MPC-24. The following table shows the results from the finite element stress analysis and from the stability calculation.

<b>Panel Buckling Results From NUREG/CR-6322</b>			
<b>Item</b>	<b>Finite Element Stress (ksi)</b>	<b>Critical Buckling Stress (ksi)</b>	<b>Factor of Safety</b>
Stress	12.585	45.32	3.601

For a stainless steel member under an accident condition load, the recommended safety factor is 2.12. We see that the calculated safety factor exceeds this value; therefore, we have independently confirmed the stability predictions of the large deflection analysis based on classical plate stability analysis by employing a simplified method.

Stability of the basket panels, under longitudinal deceleration loading, is demonstrated in the following manner. Under 60g deceleration in Docket Number 72-1008, the axial compressive stress in the baskets were computed for the MPC-24, 68, and 32, as:

MPC-24	3,458 psi
MPC-68	3,739 psi
MPC-32	4,001 psi

For the 45g design basis decelerations for HI-STORM 100, the basket axial stresses are reduced by 25%.

The above values represent the amplified weight, including the nonstructural sheathing and the *neutron absorber material* ~~Boral~~, divided by the bearing area resisting axial movement of the basket. To demonstrate that elastic instability is not a concern, the buckling stress for an MPC-24 flat panel is computed.

For elastic stability, Reference [3.4.8] provides the formula for critical axial stress as

$$\sigma_{cr} = \frac{4 \pi^2 E}{12 (1 - \nu^2)} \left( \frac{T}{W} \right)^2$$

where T is the panel thickness and W is the width of the panel, E is the Young's Modulus at the metal temperature and  $\nu$  is the metal Poisson's Ratio. The following table summarizes the calculation for the critical buckling stress using the formula given above:

<b>Elastic Stability Result for a Flat Panel</b>	
Reference Temperature	725 degrees F
T (MPC-24)	5/16 inch
W	10.777 inch
E	24,600,000 psi
Critical Axial Stress	74,781 psi

It is noted the critical axial stress is an order of magnitude greater than the computed basket axial stress reported in the foregoing and demonstrates that elastic stability under longitudinal deceleration load is not a concern for any of the fuel basket configurations.

b. Yielding

The safety factor against yielding of the basket under longitudinal compressive stress from a design basis inertial loading is given, using the results for the MPC-32, by

$$SF = 17,100/4,001 = 4.274$$

Therefore, plastic deformation of the fuel basket under design basis deceleration is not credible.

3.4.4.3.1.4 MPC Baseplate Analysis (Load Case E2)

A bounding analysis is performed in the HI-STAR 100 FSAR (Docket Number 72-1008, Appendix 3.I) to evaluate the stresses in the MPC baseplate during the handling of a loaded MPC. The stresses in the MPC baseplate calculated in that appendix are compared to Level A stress limits and remain unchanged whether the overpack is HI-STAR 100, HI-STORM 100, or HI-TRAC. Therefore, no new analysis is needed. We have reported results for this region in Subsection 3.4.3 where an evaluation has been performed for stresses under three times the supported load.

#### 3.4.4.3.1.5 Analysis of the MPC Top Closure (Load Case E2)

The FSAR for the HI-STAR 100 System (Docket Number 72-1008, Appendix 3.E) contains stress analysis of the MPC top closure during lifting. Loadings in that analysis are also valid for the HI-STORM 100 System. *From Table 2.2.1, the off-normal design internal pressure is 110 psi, or ten percent greater than the normal design pressure. Whereas Level A service limits are used to establish allowables for the normal design pressure, Level B service limits are used for off-normal loads. Since Subsection NB of the ASME Code permits an identical 10% increase in allowable stress intensity values for primary stress intensities generated by Level B Service Loadings, it stands to reason that the safety factors reported for normal pressure are also valid for the case of off-normal design internal pressure.*

#### 3.4.4.3.1.6 Structural Analysis of the Fuel Support Spacers (Load Case E3.a)

Upper and lower fuel support spacers are utilized to position the active fuel region of the spent nuclear fuel within the poisoned region of the fuel basket. It is necessary to ensure that the spacers will continue to maintain their structural integrity after an accident event. Ensuring structural integrity implies that the spacer will not buckle under the maximum compressive load, and that the maximum compressive stress will not exceed the compressive strength of the spacer material (Alloy X). Detailed calculations in Docket Number 72-1008, Appendix 3.J, demonstrate that large structural margins in the fuel spacers are available for the entire range of spacer lengths which may be used in HI-STORM 100 applications (for the various acceptable fuel types). The calculations for the HI-STORM 100 45g load are bounded by those for the HI-STAR 100 60g load.

#### 3.4.4.3.1.7 External Pressure (Load Case E1.b, Table 3.1.4)

~~The Design external pressure for is applied to the MPC is zero psi model. The outer surface of the MPC shell is conservatively subject to a net external pressure of 2 psi. The magnitude of the external pressure applied to the model is taken from Table 2.2.1. The methodology for a~~ Analysis of the MPC under this external pressure is provided in the HI-STAR 100 FSAR Docket Number 72-1008. *Using the identical methodology with input loads and decelerations appropriate to the HI-STORM, safety factors > 1.0 are obtained for all relevant load cases. (Appendix 3.H) and therefore, is not repeated here.*

#### 3.4.4.3.1.8 Miscellaneous MPC Structural Evaluations

*Calculations are performed to determine the minimum fuel basket weld size, the capacity of the sheathing welds, the stresses in the MPC cover plates, and the stresses in the fuel basket angle supports. The following paragraphs briefly describe each of these evaluations.*

*The fillet welds in the fuel basket honeycomb are made by an autogenous operation that has been shown to produce highly consistent and porosity free weld lines. However, Subsection NG of the ASME Code permits only 40% quality credit on double fillet welds which can be only visually examined (Table NG-3352-1). Subsection NG, however, fails to provide a specific stress limit on such fillet welds. In the absence of a Code mandated limit, Holtec International's standard design procedure requires that the weld section possess as much load resistance*

capability as the parent metal section. Since the loading on the honeycomb panels is essentially that of section bending, it is possible to develop a closed form expression for the required weld throat thickness “t” corresponding to panel thickness “h”.

The sheathing is welded to the fuel basket cell walls to protect and position the neutron absorber material. Force equilibrium relationships are used to demonstrate that the sheathing weld is adequate to support a 45g deceleration load applied vertically and horizontally to the sheathing and the confined neutron absorber material. The analysis assumes that the weld is continuous and then modifies the results to reflect the amplification due to intermittent welding.

The MPC cover plates are welded to the MPC lid during loading operations. The cover plates are part of the confinement boundary for the MPC. No credit is taken for the pressure retaining abilities of the quick disconnect couplings for the MPC vent and drain. Therefore, the MPC cover plates must meet ASME Code, Section III, Subsection NB limits for normal, off-normal, and accident conditions. Conservatively, the accident condition pressure loading is applied, and it is demonstrated that the Level A limits for Subsection NB are met.

The fuel basket internal to the MPC canister is supported by a combination of angle fuel basket supports and flat plate or solid bar fuel basket supports. These fuel basket supports are subject to significant load only when a lateral acceleration is applied to the fuel basket and the contained fuel. The quasi-static finite element analyses of the MPC's, under lateral inertia loading, focused on the structural details of the fuel basket and the MPC shell. Basket supports were modeled in less detail, which served only to properly model the load transfer path between fuel basket and canister. Safety factors reported for the fuel basket supports from the finite element analyses, are overly conservative, and do not reflect available capacity of the fuel basket angle support. A strength of materials analysis of the fuel basket angle supports is performed to complement the finite element results. The weld stresses are computed at the support-to-shell interface, and membrane and bending stresses in the basket support angle plate itself. Using this strength of materials approach, we demonstrate that the safety factors for the fuel basket angle supports are larger than indicated by the finite element analysis.

The results of these evaluations are summarized in the tables below.

<b>Minimum Weld Sizes for Fuel Baskets</b>			
<b>Basket Type</b>	<b>Panel Thickness (h), in</b>	<b>t/h Ratio</b>	<b>Minimum Weld Size (t), in</b>
MPC-24	5/16	0.57	0.178
MPC-68	1/4	0.516	0.129
MPC-32	9/32	0.57	0.160
MPC-24E	5/16	0.455	0.142

<i>Miscellaneous Stress Results for MPC</i>			
<i>Item</i>	<i>Stress (ksi)</i>	<i>Allowable Stress (ksi)</i>	<i>Safety Factor</i>
<i>Shear Stress in Sheathing Weld</i>	<i>2.968</i>	<i>27.93</i>	<i>9.41</i>
<i>Bending Stress in MPC Cover Plate</i>	<i>17.60</i>	<i>24.425*0.967</i>	<i>1.349</i>
<i>Shear Stress in MPC Cover Plate Weld</i>	<i>3.145</i>	<i>18.99*0.967</i>	<i>5.846.04</i>
<i>Shear Stress in Fuel Basket Support Weld</i>	<i>4.711</i>	<i>9.408</i>	<i>2.00</i>
<i>Combined Stress in Fuel Basket Support Plates</i>	<i>32.393</i>	<i>59.1</i>	<i>1.82</i>

*Note: 0.967 reflects increase in MPC shell design temperature to 500 deg. F*

#### *3.4.4.3.1.9 Structural Integrity of Damaged Fuel Containers*

*The damaged fuel containers or canisters (DFCs) to be deployed in the HI-STAR 100 System transport package have been evaluated to demonstrate that the containers are structurally adequate to support the mechanical loads postulated during normal lifting operations, while in long-term storage, and during a hypothetical end drop. The evaluations address the following damaged/failed fuel containers for transportation in the HI-STAR 100 System:*

- Holtec-designed MPC-24E (PWR) DFC*
- Holtec-designed MPC-68 (BWR) DFC*
- Transnuclear-designed DFC for Dresden Unit 1 fuel*
- Transnuclear-designed Thoria Rod Canister for Dresden Unit 1*

*The structural load path in each of the analyzed containers is evaluated using basic strength of materials formulations. The various structural components are modeled as axial or bending members and their stresses are computed. Depending on the particular DFC, the load path includes components such as the container sleeve and collar, various weld configurations, load tabs, closure components and lifting bolts. Axial plus bending stresses are computed, together with applicable bearing stresses and weld stresses. Comparisons are then made with the appropriate allowable strengths at temperature. Input data for all DFCs comes from the applicable drawings. The design temperature for lifting evaluations is set at 150°F (since the DFC is in the spent fuel pool). The design temperature for accident conditions is set at 725°F.*

*The upper closure assembly must meet the requirements set forth for special lifting devices used in nuclear applications [3.1.2]. The remaining components of the damaged fuel container are governed by the stress limits of the ASME Code Section III, Subsection NG [3.4.10] and Section III, Appendix F [3.4.3], as applicable.*

The following table presents the minimum safety factors, from all of the stress computations, for each of the above listed DFCs.

<b>DFC Type</b>	<b>Loading Condition – DFC Component</b>	<b>Calculated Stress (ksi)</b>	<b>Allowable Stress (ksi)</b>	<b>Safety Factor = (Allowable Stress) / (Calculated Stress)</b>	<b>Remarks</b>
<i>Holtec-designed MPC-24E (PWR) DFC</i>	<i>Normal Lift – Lifting Bolt</i>	24.99	27.00	1.08	<i>ANSI N14-6 stress limit</i>
	<i>60g End Drop – Baseplate-to-Container Sleeve Welds</i>	3.95	26.59	6.73	<i>ASME Level D stress limit</i>
<i>Holtec-designed MPC-68 (BWR) DFC</i>	<i>Normal Lift – Lifting Bolt-to-Top Plate Weld</i>	5.80	12.00	2.07	<i>ASME Level A stress limit</i>
	<i>60g End Drop – Baseplate-to-Container Sleeve Welds</i>	1.59	26.59	16.7	<i>ASME Level D stress limit</i>
<i>Transnuclear-designed DFC for Dresden Unit 1</i>	<i>Normal Lift – Lid Frame Assembly</i>	0.527	4.583	8.70	<i>Bearing stress</i>
	<i>60g End Drop – Bottom Assembly</i>	12.32	37.92	3.08	<i>ASME Level D stress limit</i>
<i>Transnuclear-designed Thoria Rod Canister for Dresden Unit 1</i>	<i>Normal Lift – Lid Frame Assembly</i>	0.373	4.583	12.3	<i>Bearing stress</i>
	<i>60g End Drop – Bottom Assembly</i>	8.73	37.92	4.34	<i>ASME Level D stress limit</i>

The table above demonstrates that the DFCs are structurally adequate to support the mechanical loads postulated during normal lifting operations and during a hypothetical end drop. Moreover, since the HI-STAR design basis handling accident bounds the corresponding load for HI-STORM (60g vs. 45g), the DFCs can be carried safely in both the HI-STAR and HI-STORM Systems.

#### 3.4.4.3.2 HI-STORM 100 Storage Overpack Stress Calculations

The structural functions of the storage overpack are stated in Section 3.1. The analyses presented here demonstrate the ability of components of the HI-STORM 100 storage overpack to perform their structural functions in the storage mode. Load Cases considered are given in Table 3.1.5. The nomenclature used to identify the load cases (Load Case Identifier) considered is also given in Table 3.1.5.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability of the MPC from the storage overpack. Results obtained using the HI-STORM 100 configuration are identical to or bound results for the HI-STORM 100S configuration.

3.4.4.3.2.1 HI-STORM 100 Compression Under the Static Load of a Fully Loaded HI-TRAC Positioned on the Top of HI-STORM 100 (Load Case 01 in Table 3.1.5)

During the loading of HI-STORM 100, a HI-TRAC transfer cask with a fully loaded MPC may be placed on the top of a HI-STORM 100 storage overpack. During this operation, the HI-TRAC may be held by a single-failure-proof lifting device so a handling accident is not credible. The HI-STORM 100 storage overpack must, however, possess the compression capacity to support the additional dead load. The following analysis provides the necessary structural integrity demonstration; results for the HI-STORM 100 overpack are equal to or bound those for the HI-STORM 100S.

Define the following quantities for analysis purposes:

$W_{HT}$  = Bounding weight of HI-TRAC 125D (loaded w/ MPC-32) = 233,000 lb (Table 3.2.2)

$W_{MD}$  = Weight of mating device = 15,000 lb

$W_{TOTAL} = W_{HT} + W_{MD} = 248,000$  lb

The total weight of the HI-TRAC 125D plus the mating device is greater than the weight of a loaded HI-TRAC 125 with the transfer lid. Therefore, the following calculations use the weight for the HI-TRAC 125D as input.

The dimensions of the compression components of HI-STORM 100 are as follows:

outer diameter of outer shell =	$D_o = 132.5''$
thickness of outer shell =	$t_o = 0.75''$
outer diameter of inner shell =	$D_i = 76''$
thickness of inner shell =	$t_i = 1.25''$
thickness of radial ribs =	$t_r = 0.75''$

The metal area of the outer metal shell is

$$A_o = \frac{\pi}{4} (D_o^2 - (D_o - 2 t_o)^2) = \frac{\pi}{4} (132.5^2 - 131^2) \\ = 310.43 \text{ in}^2$$

The metal area of the radial ribs is

$$A_r = 4 t_r (D_o - 2 t_o - D_i) / 2 = \frac{3}{2} (131 - 76) = 82.5 \text{ in}^2$$

The metal area of the inner shell is

$$A_i = \frac{\pi}{4} (D_i^2 - (D_i - 2t_i)^2) = \frac{\pi}{4} (76^2 - 73.5^2) \\ = 293.54 \text{ in}^2$$

There are four radial ribs that extend full length and can carry load. The concrete radial shield can also support compression load. The area of concrete available to support compressive loading is

$$A_{\text{concrete}} = \frac{\pi}{4} ((D_o - 2t_o)^2 - (D_i)^2) - A_r \\ = \frac{\pi}{4} (131^2 - 76^2) - 82.5 \text{ in}^2 \\ = (8,994 - 82.5) \text{ in}^2 = 8,859.5 \text{ in}^2$$

The areas computed above are calculated at a section below the air outlet vents. To correct the above areas for the presence of the air outlet vents (HI-STORM 100 only since HI-STORM 100S has the air outlet vents located in the lid), we note that Bill-of-Materials 1575 in Chapter 1 gives the size of the horizontal plate of the air outlet vents as:

Peripheral width =  $w = 16.5''$

Radial depth =  $d = 27.5''$  (over concrete in radial shield)

Using these values, the following final areas are obtained:

$$A_o = A_o(\text{no vent}) - 4t_o w = 260.93 \text{ sq. inch}$$

$$A_i = A_i(\text{no vent}) - 4t_i w = 211.04 \text{ sq. inch}$$

$$A_{\text{concrete}} = A_{\text{concrete}}(\text{no vent}) - 4dw = 7044.2 \text{ sq. inch}$$

The loading case is a Level A load condition. The load is apportioned to the steel and to the concrete in accordance with the values of EA for the two materials ( $E(\text{steel}) = 28,000,000 \text{ psi}$  and  $E(\text{concrete}) = 3,605,000 \text{ psi}$ ).

$$EA(\text{steel}) = 28 \times 10^6 \text{ psi} \times (260.93 + 211.04 + 82.5) \text{ in}^2 \\ = 15,525.2 \text{ lb} \times 10^6 \text{ lbs.}$$

$$EA(\text{concrete}) = 3.605 \times 10^6 \times (7044.2) \text{ in}^2 \\ = 25,394.3 \times 10^6 \text{ lb.}$$



Therefore, the total HI-TRAC load will be apportioned as follows:

$$F_{STEEL} = (15,525.2/40,919.5) \times 248,000 = 94,093.2 \text{ lb.}$$

$$F_{CONCRETE} = (25,394.3/40,919.5) \times 248,000 = 153,906.7 \text{ lb.}$$

Therefore, if the load is apportioned as above, with all load-carrying components in the path acting, the compressive stress in the steel is

If we conservatively neglect the compression load bearing capacity of concrete, then

$$\sigma_{STEEL} = \frac{248,000}{554.5} = 447.2 \text{ psi}$$

If we include the concrete, then the maximum compressive stress in the concrete is:

$$\sigma_{CONCRETE} = \frac{F_{CONCRETE}}{A_{CONCRETE}} = 21.8 \text{ psi}$$

It is clear that HI-STORM 100 storage overpack can support the dead load of a fully loaded HI-TRAC 125D and the mating device placed on top for MPC transfer into or out of the HI-STORM 100 storage overpack cavity. The calculated stresses at a cross-section through the air outlet ducts are small and give rise to large factors of safety. The metal cross-section at the base of the HI-STORM storage overpack will have a slightly larger metal area (because the width of the air-inlet ducts is smaller) but will be subject to additional dead load from the weight of the supported metal components of the HI-STORM storage overpack plus the loaded HI-TRAC weight. At the base of the storage overpack, the additional stress in the outer shell and the radial plates is due solely to the weight of the component. Based on the maximum concrete density, the additional stress in these components is computed as:

$$\Delta\sigma = (160.8 \text{ lb./cu.ft.}) \times 18.71 \text{ ft./144 sq.in./sq.ft.} = 20.9 \text{ psi}$$

This stress will be further increased by a small amount because of the material cut away by the air-inlet ducts; however, the additional stress still remains small. The inner shell, however, is subject to additional loading from the top lid of the storage overpack and from the radial shield. From the Structural Calculation Package (HI-981928)(see Subsection 3.6.4 for the reference), and from Table 3.2.1, the following weights are obtained (using the higher 100S lid weight):

HI-STORM 100S Top Lid weight < 25,500 lb.

HI-STORM 100 Inner Shell weight < 19,000 lb.

HI-STORM 100 Shield Shell weight < 11,000 lb.

Note that the shield shell was removed from the HI-STORM 100 design as of June, 2001. However, it is conservative to include the shield shell weight in the following calculations.

Using the calculated inner shell area at the top of the storage overpack for conservatism, gives the metal area of the inner shell as:

$$A_i = A_i(\text{no vent}) - 4t_i w = 211.04 \text{ sq. inch}$$

Therefore, the additional stress from the HI-STORM 100S storage overpack components, at the base of the overpack, is:

$$\Delta\sigma = 263 \text{ psi}$$

and a maximum compressive stress in the inner shell predicted as:

$$\text{Maximum stress} = 447 \text{ psi} + 263 \text{ psi} = 710 \text{ psi}$$

The safety factor at the base of the storage overpack inner shell (minimum section) is

$$\text{SF} = 17,500\text{psi}/710 \text{ psi} = 24.6$$

The preceding analysis is bounding for the 100 Ton HI-TRAC transfer cask because of the lower HI-TRAC weight.

The preceding analysis is valid for both the HI-STORM 100 and the HI-STORM 100S since the bounding lid weight has been used.

#### 3.4.4.3.2.2 HI-STORM 100 Lid Integrity Evaluation (Load Case 02.c, Table 3.1.5)

A non-mechanistic tip over of the HI-STORM 100 results in high decelerations at the top of the storage overpack. The storage overpack lid diameter is less than the storage overpack outer diameter. This ensures that the storage overpack lid does not directly strike the ground but requires analysis to demonstrate that the lid remains intact and does not separate from the body of the storage overpack. Figure 3.4.19 shows the scenario.

The HI-STORM 100 overpack has two lid designs, which rely on different mechanisms to resist separation from the overpack body. The original design relies solely on the lid studs to resist the shear and axial loads on the lid. In the new design, the bolt holes are enlarged and a shear ring is welded to the underside of the lid top plate. These changes insure that the lid studs only encounter axial (tensile) loads. The in-plane load is resisted by the shear ring as it bears against the top plate. The HI-STORM 100S has only one lid design, which utilizes a shear ring. Calculations have been performed for both HI-STORM 100 lid configurations, as well as the HI-STORM 100S lid geometry, to demonstrate that the lid can withstand a non-mechanistic tip-over.

Specifically, Appendix 3.K presents details of the HI-STORM 100 lid response to the tip-over deceleration based on the original design (i.e., no shear ring). The deceleration level for the non-mechanistic tip-over bounds all other decelerations, directed in the plane of the lid, experienced under other accident conditions such as flood or earthquake as can be demonstrated by evaluating the loads resulting from these natural phenomena events.

Appendix 3.L presents the original calculations that demonstrate that the four studs hold the storage overpack lid in place, relative to the HI-STORM 100 body, for a postulated non-mechanistic HI-STORM 100 tip-over event. It is shown that the weight of the HI-STORM 100 lid, amplified by the design basis deceleration, can be supported entirely by the shear capacity available in the four studs<sup>†</sup>. The detailed calculations in Appendix 3.L demonstrate that if only a single stud is loaded initially during a tipover (because of tolerances), the stud hole will enlarge rather than the stud fail in shear. Therefore, it is assured that all four bolts will resist the tipover load regardless of the initial position of the HI-STORM 100 lid.

The following tables summarize the limiting results obtained from the detailed analyses in Appendices 3.K and 3.L, and from the similar detailed analysis for the HI-STORM 100 lid with shear ring and for the HI-STORM 100S(243). The results for the HI-STORM 100S(243) bound the results for the shorter HI-STORM 100S(232).

<b>HI-STORM 100 Top Lid Integrity (No Shear Ring)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Shell-Lid Top Plate Weld Shear Stress	6.733	29.4	4.367
Lid Shell-Lid Top Plate Combined Stress	9.11	29.4	3.226
Attachment Bolt Shear Stress	44.82	60.9	1.359
Attachment Bolt Combined Shear and Tension Interaction at Interface with Anchor Block	-----	-----	1.21

<sup>†</sup> The tip-over event is non-mechanistic by definition since the HI-STORM 100 System is designed to preclude tip-over under all normal, off-normal, and accident conditions of storage, including extreme natural phenomena events. Thus, the tip-over event cannot be categorized as an operating or test condition as contemplated by ASME Section III, Article NCA-2141. The bolted connection between the overpack top lid and the overpack body provided by the top lid studs and nuts serves no structural function during normal or off-normal storage conditions, or for credible accident events. Therefore, the ASME Code does not apply to the construction of the HI-STORM top plate-to-overpack connection (the lid studs, nuts, and the through holes in the top plate). However, for conservatism, the stress limits from ASME III, Subsection NF are used for the analysis of the lid bolts in Appendix 3.L.

<b>HI-STORM 100 Top Lid Integrity (With Shear Ring)</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Top Plate-to-Lid Shell Weld Combined Stress	7.336	29.4	4.007
Shield Block Shells-to-Lid Top Plate Weld Combined Stress	1.768	29.4	16.63
Attachment Bolt Tensile Stress	28.02	107.13	3.823
Shear Ring-to-Lid Top Plate Weld Stress	32.11	40.39	1.258
Shear Ring Bearing Stress	25.43	63.0	2.477
Top Plate-to-Outer Shell Weld Stress	35.61	40.39	1.134

<b>HI-STORM 100S(243) Top Lid Integrity</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Inner and Outer Shell Weld to Base	15.98	29.4	1.840
Shield Block Shell-to-Lid Weld Shear Stress	5.821	29.4	5.051
Shield Block Shell Stress	5.975	29.4	4.921
Attachment Bolt Tensile Stress	34.04	107.13	3.147
Shear Ring-to Overpack Shell Weld Stress	30.27	42.0	1.388
Shear Ring Bearing Stress	17.63	63.0	3.573
Lid Shell Ring-to-Shear Ring Weld Stress	19.01	42.0	2.209

#### 3.4.4.3.2.3 Vertical Drop of HI-STORM 100 Storage overpack (Load Case 02.a of Table 3.1.5)

A loaded HI-STORM 100, with the top lid in place, drops vertically and impacts the ISFSI. Figure 3.4.20 illustrates the drop scenario. The regions of the structure that require detailed examination are the storage overpack top lid, the inlet vent horizontal plate, the pedestal shield, the inlet vent vertical plate, and all welds in the load path. *These components are examined for Appendix 3.M examines the Level D event of a HI-STORM 100 drop developing the design basis deceleration.*

The table provided below summarizes the results of the analyses detailed in Appendix 3.M for the weight and configuration of the HI-STORM 100. The results for the HI-STORM 100S are bounded by the results given below. Any calculation pertaining to the pedestal is bounding since the pedestal dimensions and corresponding weights are less in the HI-STORM 100S.

<b>HI-STORM 100 Load Case 02.a Evaluation</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
Lid Bottom Plate Bending Stress Intensity	6.02	59.65	9.908 <sup>†</sup>
Weld- lid bottom plate-to-lid shell	10.91	29.4	2.695
Lid Shell – Membrane Stress Intensity	1.90	39.75	20.92
Lid Top (2" thick) Plate Bending Stress Intensity	11.27	59.65	5.293*
Inner Shell –Membrane Stress Intensity	8.88	39.75	4.476
Outer Shell –Membrane Stress Intensity	3.401	39.75	11.686
Inlet Vent Horizontal Plate Bending Stress Intensity	37.14	59.65	1.606
Inlet Vent Vertical Plate Membrane Stress Intensity	10.34	39.75	3.844
Pedestal Shield – Compression	1.252	1.266	1.011
Weld – outer shell-to-baseplate	4.133	29.4	7.116
Weld – inner shell-to-baseplate	5.896	29.4	4.987
Weld-Pedestal shell-to-baseplate	4.563	29.4	6.444

<sup>†</sup> Note that ~~Appendix 3.X~~ shows that the dynamic load factor for the lid top plate is negligible and for the lid bottom plate is 1.06. This dynamic load factor has been incorporated in the above table.

\* For the HI-STORM 100S, this safety factor is conservatively evaluated in ~~Appendix 3.M~~ to be 1.625 because of increased load on the upper of the two lid plates.

~~Appendix 3.AK~~ contains an assessment of the potential for instability of the compressed inner and outer shells under the compressive loading during the drop event *has also been performed*. The methodology is from ASME Code Case N-284 (Metal Containment Shell Buckling Design Methods, Division I, Class MC (8/80)). This Code Case has been previously accepted by the NRC as an acceptable method for evaluation of stability in vessels. The results obtained are conservative in that the loading in the shells is assumed to be uniformly distributed over the entire length of the shells. In reality, the component due to the amplified weight of the shell varies from zero at the top of the shell to the maximum value at the base of the shell. It is concluded in ~~Appendix 3.AK~~ that large factors of safety exist so that elastic or plastic instability of the inner and outer shells does not provide a limiting condition. The results for the HI-STORM 100 bound similar results for the HI-STORM 100S since the total weight of the “S” configuration is decreased (see Subsection 3.2).

The results from ~~Appendix 3-M and 3-AK~~ do not show any gross regions of stress above the material yield point that would imply the potential for gross deformation of the storage overpack subsequent to the handling accident. MPC stability has been evaluated in the HI-STAR 100 FSAR for a drop event with 60g deceleration and shown to satisfy the Code Case N-284 criteria. Therefore, ready retrievability of the MPC is maintained as well as the continued performance of the HI-STORM 100 storage overpack as the primary shielding device.

#### 3.4.4.3.3 HI-TRAC Transfer Cask Stress Calculations

The structural functions of the transfer cask are stated in Section 3.1. The analyses presented here demonstrate the ability of components of the HI-TRAC transfer cask to perform their structural functions in the transfer mode. Load Cases considered are given in Table 3.1.5.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability.

##### 3.4.4.3.3.1 Analysis of Pocket Trunnions (Load Case 01 of Table 3.1.5)

The HI-TRAC 125 and HI-TRAC 100 transfer casks have pocket trunnions attached to the outer shell and to the water jacket. During the rotation of HI-TRAC from horizontal to vertical or vice versa (see Figure 3.4.18), these trunnions serve to define the axis of rotation. The HI-TRAC is also supported by the lifting trunnions during this operation. Two load conditions are considered: Level A when all four trunnions support load during the rotation; and, Level B when the hoist cable is assumed slack so that the entire load is supported by the rotation trunnions. A dynamic amplification of 15% is assumed in both cases appropriate to a low-speed operation. ~~Appendices 3-AA and 3-AI (for the HI-TRAC 125 and HI-TRAC 100, respectively) present the analysis of the pocket trunnion.~~ Figure 3.4.23 shows a free body of the trunnion and shows how the applied force and moment are assumed to be resisted by the weld group that connects the trunnion to the outer shell. Drawings 1880 (sheet 10) and 2145 (sheet 10) show the configuration. An optional construction for the HI-TRAC 100 permits the pocket trunnion base to be split to reduce the “envelope” of the HI-TRAC. For that construction, bolts and dowel pins are used to insure that the force and moment applied to the pocket trunnions are transferred properly to the body of the transfer cask. ~~The analysis Appendix 3-AI~~ also evaluates the bolts and dowel pins and demonstrates that safety factors greater than 1.0 exist for bolt loads, dowel bearing and tear-out, and dowel shear. Allowable strengths and loads are computed using applicable sections of ASME Section III, Subsection NF.

Unlike the HI-TRAC 125 and the HI-TRAC 100, the HI-TRAC 125D is designed and fabricated without pocket trunnions. An L-shaped rotation frame is used to upend and downend the HI-TRAC 125D, instead of pocket trunnions. Thus, a pocket trunnion analysis is not applicable to the HI-TRAC 125D.

The table below summarizes the results for the HI-TRAC 125 and the HI-TRAC 100 from the two appendices:

<b>Pocket Trunnion Weld Evaluation Summary</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)<sup>†</sup></b>	<b>Safety Factor</b>
HI-TRAC 125 Pocket Trunnion-Outer Shell Weld Group Stress	7.979	23.275	2.917
HI-TRAC 125 Pocket Trunnion-Water Jacket Weld Group Stress	5.927	23.275	3.9
HI-TRAC 100 Pocket Trunnion-Outer Shell Weld Group Stress	6.603	23.275	3.525
HI-TRAC 100 Pocket Trunnion-Water Jacket Weld Group Stress	5.244	23.275	4.438
HI-TRAC 100 Pocket Trunnion-Bolt Tension at Optional Split	45.23	50.07	1.107
HI-TRAC 100 Pocket Trunnion-Bearing Stress on Base Surfaces at Dowel	6.497	32.7	5.033
HI-TRAC 100 Pocket Trunnion-Tear-out Stress on Base Surfaces at Dowel	2.978	26.09	8.763
HI-TRAC 100 Pocket Trunnion-Shear Stress on Dowel Cross Section at Optional Split	29.04	37.93	1.306

<sup>†</sup> Allowable stress is reported for the Level B loading, which results in the minimum safety factor.

To provide additional information on the local stress state adjacent to the rotation trunnion, ~~Appendix 3-AA also includes~~ a new finite element analysis *is undertaken to provide* ~~providing~~ details on the state of stress in the metal structure surrounding the rotation trunnions for the HI-TRAC 125. The finite element analysis has been based on a model that includes major structural contributors from the water jacket enclosure shell panels, radial channels, end plates, outer and inner shell, and bottom flange. In the finite element analysis, the vertical trunnion load has been oriented in the direction of the HI-TRAC 125 longitudinal axis. The structural model has been confined to the region of the HI-TRAC adjacent to the rotation trunnion block; the extent of the model in the longitudinal direction has been determined by calculating the length of the “bending boundary layer” associated with a classical shell analysis. This was considered to be a sufficient length to capture

maximum shell stresses arising from the Level B (off-normal) rotation trunnion loading. Appendix 3.AA contains the results of the finite element simulations with complete graphical output showing the longitudinal and circumferential stress distribution in the inner and outer shells and in the radial channels. The local nature of the stress around the trunnion block is clearly demonstrated by the finite element graphical results.

Consistent with the requirements of ASME Section III, Subsection NF, for Class 3 components, safety factors for primary membrane stress have been computed. Primary stresses are located away from the immediate vicinity of the trunnion; although the NF Code sets no limits on primary plus secondary stresses that arise from the gross structural discontinuity immediately adjacent to the trunnion, these stresses are listed for information. The results, assembled from the results in Appendix 3.AA, are summarized in the table below for the Level B load distribution for the HI-TRAC 125.

ITEM –HI-TRAC 125	CALCULATED VALUE	ALLOWABLE VALUE
Longitudinal Stress - (ksi) (Primary Stress –Inner Shell)	-0.956	23.275
Tangential Stress (ksi) (Primary Stress - Inner Shell)	-1.501	23.275
Longitudinal Stress (ksi) (Primary Stress – Outer Shell)	-0.830	23.275
Tangential Stress (ksi) (Primary Stress - Outer Shell)	-0.436	23.275
Longitudinal Stress - (ksi) (Primary Stress – Radial Channels)	2.305	23.275
Tangential Stress (ksi) (Primary Stress - Radial Channels)	-0.631	23.275
Longitudinal Stress - (ksi) (Primary plus Secondary Stress -Inner Shell)	1.734	No Limit (34.9)*
Tangential Stress (ksi) (Primary plus Secondary Stress - Inner Shell)	-1.501	NL
Longitudinal Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	2.484	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	-2.973	NL
Longitudinal Stress - (ksi) (Primary plus Secondary Stress - Radial Channels)	-13.87	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Radial Channels)	-2.303	NL

\* The NF Code sets no limits (NL) for primary plus secondary stress (see Table 3.1.17). Nevertheless, to demonstrate the robust design with its large margins of safety, we list here, for information only, the allowable value for Primary Membrane plus Primary Bending Stress appropriate to temperatures up to 650 degrees F.



The only stress of any significance is the longitudinal stress in the radial channels. This stress occurs immediately adjacent to the trunnion block/radial channel interface and by its localized nature is identifiable as a stress arising at the gross structural discontinuity (secondary stress).

The finite element analysis has also been performed for the HI-TRAC 100 transfer cask; results are reported in Appendix 3-AI. The following table summarizes the results:

<b>ITEM –HI-TRAC 100</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Longitudinal Stress - (ksi) (Primary Stress –Inner Shell)	-0.756	23.275
Tangential Stress (ksi) (Primary Stress - Inner Shell)	-2.157	23.275
Longitudinal Stress (ksi) (Primary Stress – Outer Shell)	-0.726	23.275
Tangential Stress (ksi) (Primary Stress - Outer Shell)	-0.428	23.275
Longitudinal Stress - (ksi) (Primary Stress – Radial Channels)	2.411	23.275
Tangential Stress (ksi) (Primary Stress - Radial Channels)	-0.5305	23.275
Longitudinal Stress - (ksi) (Primary plus Secondary Stress - Inner Shell)	2.379	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Inner Shell)	-2.157	NL
Longitudinal Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	3.150	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Outer Shell)	-3.641	NL
Longitudinal Stress - (ksi) (Primary plus Secondary Stress - Radial Channels)	-15.51	NL
Tangential Stress (ksi) (Primary plus Secondary Stress - Radial Channels)	-2.294	NL

The finite element analyses of the metal structure adjacent to the trunnion block did not include the state of stress arising from the water jacket internal pressure. These stresses ~~are computed in Appendix 3.AG and~~ are conservatively computed based on a two-dimensional strip model that neglects the lower annular plate. The water jacket bending stresses ~~calculated in Appendix 3.AG~~ are summarized below:

<b>Appendix 3.AG Result for Tangential Bending Stress in Water Jacket Outer Panel from Water Pressure (including hydrostatic and inertia effects)</b>	<b>Calculated Value (ksi)</b>
HI-TRAC 125	18.41
HI-TRAC 100	22.47

To establish a minimum safety factor for the outer panels of the water jacket for the Level A condition, we must add primary membrane circumferential stress from the trunnion load analysis (~~Appendices 3.AA and 3.AI with reduction factor from Level B to Level A load~~) to primary circumferential bending stress from the water jacket bending stress (~~Appendix 3.AG~~). Then, the safety factors may be computed by comparison to the allowable limit for primary membrane plus primary bending stress. The following results are obtained:

<b>Results for Load Case 01 in Water Jacket (Load Case 01) – Level A Load</b>			
<b>Circumferential Stress in Water Jacket Outer Enclosure</b>	<b>CALCULATED VALUE (ksi)</b>	<b>ALLOWABLE VALUE (ksi)</b>	<b>SAFETY FACTOR (allowable value/calculated value)</b>
HI-TRAC 125	18.797	26.25	1.397
HI-TRAC 100	22.781	26.25	1.152

To arrive at minimum safety factors for primary membrane plus bending stress in the outer panel of the water jacket for the Level B condition, we amplify the finite element results *from the trunnion load analysis* ~~in accordance with Appendices 3.AA and 3.AI~~, add the appropriate stress from *the two-dimensional water jacket calculation* ~~Appendix 3.AG~~, and compare the results to the increased Level B allowable. The following results are obtained:

<b>Results for Load Case 01 in Water Jacket (Load Case 01) – Level B Load</b>			
<b>Circumferential Stress in Water Jacket Outer Enclosure</b>	<b>CALCULATED VALUE (ksi)</b>	<b>ALLOWABLE VALUE (ksi)</b>	<b>SAFETY FACTOR (allowable value/calculated value)</b>
HI-TRAC 125	19.041	35.0	1.84
HI-TRAC 100	23.00	35.0	1.52

All safety factors are greater than 1.0; the Level A load condition governs.

#### 3.4.4.3.3.2 Lead Slump in HI-TRAC 125 - Horizontal Drop Event (Case 02.b in Table 3.1.5)

During a side drop of the HI-TRAC 125 transfer cask, the lead shielding must be shown not to slump and cause significant amounts of shielding to be lost in the top area of the lead annulus. Slumping of the lead is not considered credible in the HI-TRAC transfer cask because of:

- a. the shape of the interacting surfaces
- b. the ovalization of the shell walls under impact
- c. the high coefficient of friction between lead and steel
- d. The inertia force from the MPC inside the HI-TRAC will compress the inner shell at the impact location and locally “pinch” the annulus that contains the lead; this opposes the tendency for the lead to slump and open up the annulus at the impact location.

Direct contact of the outer shell of the HI-TRAC with the ISFSI pad is not credible since there is a water jacket that surrounds the outer shell. The water jacket metal shell will experience most of the direct impact. Nevertheless, to conservatively analyze the lead slump scenario, it is assumed that there is no water jacket, the impact occurs far from either end of the HI-TRAC so as to ignore any strengthening of the structure due to end effects, the impact occurs directly on the outer shell of the HI-TRAC, and the contact force between HI-TRAC and the MPC is ignored. All of these assumptions are conservative in that their imposition magnifies any tendency for the lead to slump.

To confirm that lead slump is not credible, a finite element analysis of the lead slump problem, incorporating the conservatisms listed above, during a postulated HI-TRAC 125 horizontal drop (see Figure 3.4.22) is carried out. ~~Details of the analysis (finite element model and plotted results) are presented in Appendix 3.F.~~ The HI-TRAC 125 cask body modeled consists only of an inner steel shell, an outer steel shell, and a thick lead annulus shield contained between the inner and outer shell. A unit length of HI-TRAC is modeled and the contact at the lead/steel interface is modeled as a compression-only interface. Interface frictional forces are conservatively neglected. As the HI-TRAC 125 has a greater lead thickness, analysis of the HI-TRAC 125 is considered to bound the HI-TRAC-100425. Furthermore, since there are no differences between the HI-TRAC 125 and the HI-TRAC 125D with respect to the finite element model, the results are valid for both 125-Ton transfer casks.

The analysis is performed in two parts:

First, to maximize the potential for lead/steel separation, the shells are ignored and the gap elements grounded. This has the same effect as assuming the shells to be rigid and maximizes the potential and magnitude of any separation at the lead/steel interface (and subsequent slump). This also maximizes the contact forces at the portion of the interface that continues to have compression forces developed. The lead annulus is subjected to a 45g deceleration and the deformation, stress field, and interface force solution developed. This solution establishes a conservative result for the movement of the lead relative to the metal shells.

In the second part of the analysis, the lead is removed and replaced by the conservative (high) interface forces from the first part of the analysis. These interface forces, together with the 45g deceleration-induced inertia forces from the shell self weight are used to obtain a solution for the stress and deformation field in the inner and outer metal shells.

The results of the analysis described in Appendix 3.F, are as follows:

- a. The maximum predicted lead slump at a location 180 degrees from the impact point is 0.1". This gap decreases gradually to 0.0" after approximately 25 degrees from the vertical axis. It is shown in Appendix 3.F that the decrease in the diameter of the inner shell of the transfer cask (in the direction of the deceleration) is approximately 0.00054". This demonstrates that ovalization of the HI-TRAC shells does not occur. Therefore, the lead shielding deformation is confined to a local region with negligible deformation of the confining shells.
- b. The stress intensity distribution in the shells demonstrates that high stresses are concentrated, as anticipated, only near the assumed point of impact with the ISFSI pad. The value of the maximum stress intensity (51,000 psi) remains below the allowable stress intensity for primary membrane plus primary bending for a Level D event (58,700 psi). Thus, the steel shells continue to perform their function and contain the lead. The stress distribution, obtained using the conservatively large interface forces, demonstrates that permanent deformation could occur only in a localized region near the impact point. Since the "real" problem precludes direct impact with the outer shell, the predicted local yielding is simply a result of the conservatism imposed in the model.

It is concluded that a finite element analysis of the lead slump under a 45g deceleration in a side drop clearly indicates that there is no appreciable change in configuration of the lead shielding and no overstress of the metal shell structure. Therefore, retrievability of the MPC is not compromised and the HI-TRAC transfer cask continues to provide shielding.

#### 3.4.4.3.3.3 HI-TRAC Lid Stress Analysis During HI-TRAC Drop Accident (Load Case 02.b in Table 3.1.5)

Appendix 3.AD presents the stress in the HI-TRAC 125 transfer lid is analyzed stress analysis when the lid is subject to the deceleration loads of a side drop. Figure 3.4.22 is a sketch of the scenario. The analysis shows it is shown in Appendix 3.AD that the cask body, under a deceleration of 45g's, will not separate from the transfer lid during the postulated side drop. This event is considered a Level D event in the ASME parlance.

The bolts that act as doorstops to prevent opening of the doors are also checked in this appendix for their load capacity. It is required that sufficient shear capacity exists to prevent both doors from opening and exposing the MPC.

The only difference between the HI-TRAC 100 and the HI-TRAC 125 transfer lid doors is that the HI-TRAC 100 has less lead and has no middle steel plate. ~~Appendix 3.AJ presents analyses A similar analysis of to Appendix 3.AD for the HI-TRAC 100 and shows that all safety factors are greater than 1.0. The table given below summarizes the results for both units work in Appendices 3.AD and 3.AJ:~~

<b>Transfer Lid Attachment Integrity Under Side Drop</b>			
<b>Item – Shear Capacity</b>	<b>Value (kip) or (ksi)</b>	<b>Capacity (kip) or (ksi)</b>	<b>Safety Factor= Capacity/Value</b>
HI-TRAC 125 Attachment (kip)	1,272.0	1,770.0	1.392
HI-TRAC 125 Door Lock Bolts (ksi)	20.24	48.3	2.387
HI-TRAC 100 Attachment (kip)	1,129.0	1,729.0	1.532
HI-TRAC 100 Door Lock Bolts (ksi)	13.81	48.3	3.497

All safety factors are greater than 1.0 and are based on actual interface loads. ~~The actual interface load for both transfer casks is computed in Appendix 3.AN.~~ For the HI-TRAC 125 and the HI-TRAC 100, the interface load (primary impact at transfer lid) computed from the handling accident analysis is bounded by the values given below:

<b>BOUNDING INTERFACE LOADS COMPUTED FROM HANDLING ACCIDENT ANALYSES</b>	
<b>Item</b>	<b>Bounding Value from Appendix 3.AN (kip)</b>
HI-TRAC 125	1,300
HI-TRAC 100	1,150

The HI-TRAC 125D transfer cask does not utilize a transfer lid. Instead, the MPC is transferred to or from a storage overpack using the HI-TRAC pool lid and a special mating device. Therefore, an analysis is performed to demonstrate that the pool lid will not separate from the cask body during the postulated side drop. The results of this analysis are summarized in the following table.

<b>HI-TRAC 125D Pool Lid Attachment Integrity Under Side Drop</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Limit</b>	<b>Safety Factor</b>
Lateral Shear Force (kips)	562.5	1083	1.925
Maximum Bolt Tensile Stress (ksi)	11.41	116.4	10.20
Combined Tension and Shear Interaction	0.279	1.00	3.58

#### 3.4.4.3.3.4 Stress Analysis of the HI-TRAC Water Jacket (Load Case 03 in Table 3.1.5)

The water jacket is assumed subject to internal pressure from pressurized water and gravity water head. Calculations *are performed for the HI-TRAC 125, the HI-TRAC 125D, and the HI-TRAC 100* to determine the water jacket stress under internal pressure plus hydrostatic load ~~are performed in Appendix 3.AG for the HI-TRAC 125 and the HI-TRAC 100~~. Results are obtained for the water jacket configuration and the connecting welds for *all both* HI-TRAC transfer casks. The table below summarizes the results of the *analyse* analysis performed in Appendix 3.AG, as well as the results of similar calculations for the HI-TRAC 125D.

<b>Water Jacket Stress Evaluation</b>			
<b>Item</b>	<b>Value (ksi)</b>	<b>Allowable (ksi)</b>	<b>Safety Factor</b>
HI-TRAC 125 Water Jacket Enclosure Shell Panel Bending Stress	18.41	26.25	1.426
HI-TRAC 100 Water Jacket Enclosure Shell Panel Bending Stress	22.47	26.25	1.168
HI-TRAC 125 Water Jacket Bottom Flange Bending Stress	18.3	26.25	1.434
HI-TRAC 100 Water Jacket Bottom Flange Bending Stress	16.92	26.25	1.551
HI-TRAC 125 Weld Stress - Enclosure Panel Single Fillet Weld	2.22	21.0	9.454
HI-TRAC 100 Weld Stress – Enclosure Panel Single Fillet Weld	1.841	21.0	11.408
HI-TRAC 125 Weld Stress – Bottom Flange to Outer Shell Double Fillet Weld	14.79	21.0	1.42
HI-TRAC 125 - Enclosure Panel Direct Stress	1.571	17.5	11.142
HI-TRAC 100 - Enclosure Panel Direct Stress	1.736	17.5	10.84
HI-TRAC 125D Water Jacket Bottom Flange Bending Stress	18.88	26.25	1.39
HI-TRAC 125D Water Jacket Enclosure Shell Panel Bending Stress	10.80	26.25	2.43
HI-TRAC 125D Weld Stress – Enclosure Panel to Radial Rib Plug Welds	1.093	17.5	16.01
HI-TRAC 125D Weld Stress – Bottom Flange to Outer Shell Single Fillet Weld	3.133	21.0	6.70

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

~~Appendix 3.AH~~ examines the potential of top lid separation under a 45g deceleration side drop event *requires examination*. 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 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. The table below provides the results of the bounding analysis.

<b>HI-TRAC Top Lid Separation Analysis</b>			
<b>Item</b>	<b>Value</b>	<b>Capacity</b>	<b>Safety Factor= Capacity/Value</b>
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

Table 3.4.6 provides summary data extracted from the numerical analysis results for the fuel basket, enclosure vessel, and fuel basket supports based on the design basis deceleration. The results presented in Table 3.4.6 do not include any dynamic amplification due to internal elasticity of the structure (i.e., local inertia effects). ~~Appendix 3.X~~ *Calculations* suggests that a uniform conservative dynamic amplifier 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 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 basket. For example:

- i. The section properties of longitudinal fillet welds that attach contiguous cell walls to each other are completely neglected in the finite element model (Figure 3.4.7). The fillet welds strengthen the cell wall section modulus at the very locations where maximum stresses develop.
- ii. The radial gaps at the fuel basket-MPC shell and at the MPC shell-storage overpack interface are explicitly modeled. As the applied loading is incrementally increased, the MPC shell and fuel basket deform until a "rigid" backing surface of the storage overpack is contacted, making further unlimited deformation under lateral loading impossible. Therefore, some portion of the fuel basket and enclosure vessel (EV) stress has the characteristics of secondary stresses (which by definition, are self-limited by deformation in the structure to achieve compatibility). For conservativeness in the incremental analysis, we make no distinction between deformation controlled (secondary) stress and load controlled (primary) stress in the stress categorization of the MPC-24, 32, and 68 fuel baskets. We treat all stresses, regardless of their origin, as primary stresses. Such a conservative interpretation of the Code has a direct (adverse) effect on the computed safety factors. As noted earlier, the results for the MPC-24E are properly based only on primary stresses to illustrate the conservatism in the reporting of results for the MPC-24, 32, and 68 baskets.
- iii. A uniform pressure simulates the SNF inertia loading on the cell panels, which is a most conservative approach for incorporating the SNF/cell wall structure interaction.

The above assumptions act to depress the computed values of factors of safety in the fuel basket finite element analysis and render conservative results.



The reported factors of safety do not include the effect of dynamic load amplifiers. ~~As noted in Appendices 3.A and 3.X,~~ The duration of impact and the predominant natural frequency of the basket panels under drop events result in the dynamic load factors that do not exceed 1.08. Therefore, since all reported factors of safety are greater than the DLF, the MPC is structurally adequate for its intended functions.

Tables 3.4.7 and 3.4.8 report stress intensities and safety factors for the confinement boundary subject to internal pressure alone and internal pressure plus the normal operating condition temperature with the most severe thermal gradient. The final values for safety factors in the various locations of the confinement boundary provide assurance that the MPC enclosure vessel is a robust pressure vessel.

#### 3.4.4.4.2 Storage Overpack and HI-TRAC

The result from analyses of the storage overpack and the HI-TRAC transfer cask is shown in Table 3.4.5. The location of each result is indicated in the table. Safety factors for lifting operations where three times the lifted load is applied are reported in Section 3.4.3.

The table shows that all allowable stresses are much greater than their associated calculated stresses and that safety factors are above the limit of 1.0.

#### 3.4.4.5 Elastic Stability Considerations

##### 3.4.4.5.1 MPC Elastic Stability

Stability calculations for the MPC have been carried out in the HI-STAR 100 FSAR, Docket Number 72-1008, ~~Appendix 3.H. The calculations in that submittal bound calculations for the MPC in HI-STAR 100 since all loadings are identical except for the peak deceleration under accident events, which has been reduced from 60g's to 45g's. Using the identical methodology with input loads and decelerations appropriate to the HI-STAR, safety factors > 1.0 are obtained for all relevant load cases. Note that for HI-STAR, the design external pressure differential is reduced to 0.0 psi, and the peak deceleration under accident events is reduced from 60g's (HI-STAR) to 45g's.~~

##### 3.4.4.5.2 HI-STAR 100 Storage Overpack Elastic Stability

HI-STAR 100 (and 100S) storage overpack shell buckling is not a credible scenario since the two steel shells plus the entire radial shielding act to resist vertical compressive loading. Subsection 3.4.4.3.2.3 develops values for compressive stress in the steel shells of the storage overpack. Because of the low value for compressive stress coupled with the fact that the concrete shielding backs the steel shells, we can conclude that instability is unlikely. Note that the entire weight of the storage overpack can also be supported by the concrete shielding acting in compression. Therefore, in the unlikely event that a stability limit in the steel was approached, the load would simply shift to the massive concrete shielding. Notwithstanding the above comments, stability analyses of the storage overpack have been performed for bounding cases of longitudinal compressive stress with nominal circumferential compressive stress and for bounding circumferential compressive stress with nominal axial compressive stress. This latter case is for a bounding all-around external pressure

on the HI-STORM 100 outer shell. The latter case is listed as Load Case 05 in Table 3.1.5 and is performed to demonstrate that explosions or other environmental events that could lead to an all-around external pressure on the outer shell do not cause a buckling instability. ASME Code Case N-284, a methodology accepted by the NRC, has been used for this analysis. ~~Appendix 3-AK reports results of all stability analyses performed in support of this FSAR. In that appendix, the storage overpack shells are examined individually assuming that the four radial plates provide circumferential support against a buckling deformation mode. The analysis of the storage overpack outer shell for a bounding external pressure of~~

$$p_{\text{ext}} = 30 \text{ psi}$$

that, together with a nominal compressive axial load that bounds the dead weight load at the base of the outer shell, gives a safety factor against an instability of ~~(see Load Case 3 in Appendix 3-AK):~~

$$\text{Safety Factor} = (1/0.466) \times 1.34 = 2.88$$

The factor 1.34 is included in the above result since the analysis methodology of Code Case N-284 builds in this factor for a stability analysis for an accident condition.

The external pressure for the overpack stability considered here significantly bounds the short-time 10 psi differential pressure (between outer shell and internal annulus) specified in Table 2.2.1.

The same postulated external pressure condition can also act on the HI-TRAC during movement from the plant to the ISFSI pad. In this case, the lead shielding acts as a backing for the outer shell of the HI-TRAC transfer cask just as the concrete does for the storage overpack. The water jacket metal structure provides considerable additional structural support to the extent that it is reasonable to state that instability under external pressure is not credible. If it is assumed that the all-around water jacket support is equivalent to the four locations of radial support provided in the storage overpack, then it is clear that the instability result for the storage overpack bounds the results for the HI-TRAC transfer cask. This occurs because the R/t ratio (mean radius-to-wall thickness) of the HI-TRAC outer shell is less than the corresponding ratio for the HI-STORM storage overpack. Therefore, no HI-TRAC analysis is performed ~~in Appendix 3-AK.~~

### 3.4.5 Cold

A discussion of the resistance to failure due to brittle fracture is provided in Subsection 3.1.2.3.

The value of the ambient temperature has two principal effects on the HI-STORM 100 System, namely:

- i. The steady-state temperature of all material points in the cask system will go up or down by the amount of change in the ambient temperature.
- ii. As the ambient temperature drops, the absolute temperature of the contained helium will drop accordingly, producing a proportional reduction in the internal pressure in accordance with the Ideal Gas Law.

In other words, the temperature gradients in the system under steady-state conditions will remain the same regardless of the value of the ambient temperature. The internal pressure, on the other hand, will decline with the lowering of the ambient temperature. Since the stresses under normal storage condition arise principally from pressure and thermal gradients, it follows that the stress field in the MPC under -40 degree F ambient would be smaller than the "heat" condition of storage, treated in the preceding subsection. Additionally, the allowable stress limits tend to increase as the component temperatures decrease.

Therefore, the stress margins computed in Section 3.4.4 can be conservatively assumed to apply to the "cold" condition as well.

Finally, it can be readily shown that the HI-STORM 100 System is engineered to withstand "cold" temperatures (-40 degrees F), as set forth in the Technical Specification, without impairment of its storage function.

Unlike the MPC, the HI-STORM 100 storage overpack is an open structure; it contains no pressure. Its stress field is unaffected by the ambient temperature, unless low temperatures produce brittle fracture due to the small stresses which develop from self-weight of the structure and from the minute difference in the thermal expansion coefficients in the constituent parts of the equipment (steel and concrete). To prevent brittle fracture, all steel material in HI-STORM 100 is qualified by impact testing as set forth in the ASME Code (Table 3.1.18).

The structural material used in the MPC (Alloy X) is recognized to be completely immune from brittle fracture in the ASME Codes.

As no liquids are included in the HI-STORM 100 storage overpack design, loads due to expansion of freezing liquids are not considered. The HI-TRAC transfer cask utilizes demineralized water in the water jacket. However, the specified lowest service temperature for the HI-TRAC is 0 degrees F and a 25% ethylene glycol solution is required for the temperatures from 0 degrees F to 32 degrees F. Therefore, loads due to expansion of freezing liquids are not considered.

There is one condition, however, that does require examination to insure ready retrievability of the fuel. Under a postulated loading of an MPC from a HI-TRAC transfer cask into a cold HI-STORM 100 storage overpack, it must be demonstrated that sufficient clearances are available to preclude interference when the "hot" MPC is inserted into a "cold" storage overpack. To this end, an *bounding* analysis for free thermal expansions under cold conditions of storage has been performed in *Subsection 4.4.5 Appendix 3.AF*, wherein the MPC shell is postulated at its maximum design basis temperature and the thermal expansion of the overpack is ignored. The storage overpack is assumed to have been uniformly cooled to 0 degrees F from its normal assembly temperature (assumed as 70 degrees F in all analyses). The MPC is assumed to have the temperature distribution associated with being contained within a HI-TRAC transfer cask. For additional conservatism in the analysis, the MPC temperatures for the "hot condition of storage" (100 degrees F ambient) in a HI-TRAC are used to maximize the radial and axial growth of the loaded MPC. These MPC temperatures are available in Appendix 3.I. The results from the evaluation of free thermal expansion described above and carried out in detail in Appendix 3.AF for this "cold condition of transfer" are summarized in

Subsection 4.4.5, the table below: The final radial clearance (greater than 0.25” radial) is sufficient to preclude jamming of the MPC upon insertion into a cold HI-STORM 100 storage overpack.

<b>THERMOELASTIC DISPLACEMENTS IN THE HOT MPC AND COLD HI-STORM STORAGE OVERPACK UNDER COLD TEMPERATURE TRANSFER CONDITION</b>				
<b>HOT CANISTER – COLD HI-STORM</b>				
	<b>Radial Direction (in.)</b>		<b>Axial Direction (in.)</b>	
<b>Unit</b>	<b>Initial Clearance</b>	<b>Final Clearance</b>	<b>Initial Clearance</b>	<b>Final Clearance</b>
MPC (worst case)	0.5	0.364	1.0	0.24

The final radial clearance (greater than 0.25” radial) is sufficient to preclude jamming of the MPC upon insertion into a cold HI-STORM 100 storage overpack.

### 3.4.6 HI-STORM 100 Kinematic Stability under Flood Condition (Load Case A in Table 3.1.1)

The flood condition subjects the HI-STORM 100 System to external pressure, together with a horizontal load due to water velocity. Because the HI-STORM 100 storage overpack is equipped with ventilation openings, the hydrostatic pressure from flood submergence acts only on the MPC. As stated in subsection 3.1.2.1.1.3, the design external pressure for the MPC bounds the hydrostatic pressure from flood submergence. Subsection 3.4.4.5.2 has reported a positive safety factor against instability from external pressure in excess of that expected from a complete submergence in a flood. The analysis performed below is also valid for the HI-STORM 100S.

The water velocity associated with flood produces a horizontal drag force, which may act to cause sliding or tip-over. In accordance with the provisions of ANSI/ANS 57.9, the acceptable upper bound flood velocity, V, must provide a minimum factor of safety of 1.1 against overturning and sliding. For HI-STORM 100, we set the upper bound flood velocity design basis at 15 feet/sec. Subsequent calculations conservatively assume that the flow velocity is uniform over the height of the storage overpack.

The overturning horizontal force, F, due to hydraulic drag, is given by the classical formula:

$$F = C_d A V^*$$

where:

V\* is the velocity head =  $\frac{\rho V^2}{2g}$ ; ( $\rho$  is water weight density, and g is acceleration due to gravity).

A: projected area of the HI-STORM 100 cylinder perpendicular to the fluid velocity vector.

Cd: drag coefficient

The value of Cd for flow past a cylinder at Reynolds number above 5E+05 is given as 0.5 in the literature (viz. Hoerner, Fluid Dynamics, 1965).

The drag force tending to cause HI-STORM 100's sliding is opposed by the friction force, which is given by

$$F_f = \mu K W$$

where:

$\mu$  = limiting value of the friction coefficient at the HI-STORM 100/ISFSI pad interface (conservatively taken as 0.25, although literature citations give higher values).

K = buoyancy coefficient (documented in HI-981928, Structural Calculation Package for HI-STORM 100 (see citation in Subsection 3.6.4).

W: Minimum weight of HI-STORM 100 with an empty MPC.

### Sliding Factor of Safety

The factor of safety against sliding,  $\beta_1$ , is given by

$$\beta_1 = \frac{F_f}{F} = \frac{\mu K W}{C_d A V^*}$$

It is apparent from the above equation,  $\beta$ , will be minimized if the empty weight of HI-STORM 100 is used in the above equation.

As stated previously,  $\mu = 0.25$ ,  $C_d = 0.5$ .

$V^*$  corresponding to 15 ft./sec. water velocity is 218.01 lb per sq. ft.

A = length x diameter of HI-STORM 100 = 132.5" x 231.25"/144 sq. in./sq.ft. = 212.78 sq. ft.

K = buoyancy factor = 0.64 (per calculations in HI-981928)

W = empty weight of overpack w/ lid = 270,000 lbs. (Table 3.2.1)

Substituting in the above formula for  $\beta$ , we have

$$\beta_1 = 1.86 > 1.1 \text{ (required)}$$

Since the weight of the HI-STORM 100S plus the weight of an empty MPC-32 (i.e., the lightest MPC) is greater than 270,000 lb, the above calculation is also valid for the HI-STORM 100S.

### Overturning Factor of Safety

For determining the margin of safety against overturning  $\beta_2$ , the cask is assumed to pivot about a fixed point located at the outer edge of the contact circle at the interface between HI-STORM 100 and the ISFSI. The overturning moment due to a force  $F_T$  applied at height  $H^*$  is balanced by a restoring moment from the reaction to the cask buoyant force  $KW$  acting at radius  $D/2$ .

$$F_T H^* = KW \frac{D}{2}$$

$$F_T = \frac{K W D}{2 H^*}$$

$W$  is the empty weight of the storage overpack.

We have,

$$W = 270,000 \text{ lb. (Table 3.2.1)}$$

$$H^* = 119.2" \text{ (maximum height of mass center per Table 3.2.3)}$$

$$D = 132.5" \text{ (Holtec Drawing 1495)}$$

$$K = 0.64 \text{ (calculated in HI-981928)}$$

$$F_T = 96,040 \text{ lb.}$$

$F_T$  is the horizontal drag force at incipient tip-over.

$$F = C_d A V^* = 23,194 \text{ lbs. (drag force at 15 feet/sec)}$$

The safety factor against overturning,  $\beta_2$ , is given as:

$$\beta_2 = \frac{F_T}{F} = 4.14 > 1.1 \text{ (required)}$$

This result bounds the result for the HI-STORM 100S since the calculation uses a conservative lower bound weight and a bounding height for the center of gravity.

In the next subsection, results are presented to show that the load F (equivalent to an inertial deceleration of  $F/360,000 \text{ lb} = 0.0644 \text{ g}$ 's applied to the loaded storage overpack) does not lead to large global circumferential stress or ovalization of the storage overpack that could prevent ready retrievability of the MPC. It is shown in Subsection 3.4.7 that a horizontal load equivalent to  $0.47\text{g}$ 's does not lead to circumferential stress levels and ovalization of the HI-STORM storage overpack to prevent ready retrievability of the MPC. The load used for that calculation clearly bounds the side load induced by flood.

### 3.4.7 Seismic Event and Explosion - HI-STORM 100

#### 3.4.7.1 Seismic Event (Load Case C in Table 3.1.1)

##### Overturing Analysis

The HI-STORM 100 System plus its contents may be assumed to be subject to a seismic event consisting of three orthogonal statistically independent acceleration time-histories. For the purpose of performing a conservative analysis to determine the maximum ZPA that will not cause incipient tipping, the HI-STORM 100 System is considered as a rigid body subject to a net horizontal quasi-static inertia force and a vertical quasi-static inertia force. This is consistent with the approach used in previously licensed dockets. The vertical seismic load is conservatively assumed to act in the most unfavorable direction (upwards) at the same instant. The vertical seismic load is assumed to be equal to or less than the net horizontal load with  $\epsilon$  being the ratio of vertical component to one of the horizontal components. For use in calculations, define  $D_{\text{BASE}}$  as the contact patch diameter, and  $H_{\text{CG}}$  as the height of the centroid of an empty HI-STORM 100 System (no fuel). Conservatively, assume

$$D_{\text{BASE}} = 132.5" \text{ (Drawing 1495, Sheet 1 specifies } 133.875" \text{ including overhang for welding)}$$

Tables 3.2.1 and 3.2.3 give HI-STORM 100 weight data and center-of-gravity heights.

The weights and center-of-gravity heights are reproduced here for calculation of the composite center-of-gravity height of the storage overpack together with an empty MPC.

<u>Weight (pounds)</u>	<u>C.G. Height (Inches); H</u>
Overpack - $W_o = 270,000$	116.8
MPC-24 - $W_{24} = 42,000$	$109.0 + 24 = 133.0^\dagger$
MPC-68 - $W_{68} = 39,000$	$111.5 + 24 = 135.5$
MPC-32 - $W_{32} = 36,000$	$113.2 + 24 = 137.2$
MPC-24E - $W_{24E} = 45,000$	$108.9 + 24 = 132.9$

<sup>†</sup> From Table 3.2.3, it is noted that MPC C.G. heights are measured from the base of the MPC. Therefore, the thickness of the overpack baseplate and the concrete MPC pedestal must be added to determine the height above ground.

The height of the composite centroid,  $H_{CG}$ , is determined from the equation

$$H_{cg} = \frac{W_o \times 116.8 + W_{MPC} \times H}{W_o + W_{MPC}}$$

Performing the calculations for all of the MPCs gives the following results:

$H_{cg}$ (inches)	
MPC-24 with storage overpack	118.98
MPC-68 with storage overpack	119.16
MPC-32 with storage overpack	119.20
MPC-24E with storage overpack	119.10

A conservative overturning stability limit is achieved by using the largest value of  $H_{CG}$  (call it  $H$ ) from the above. Because the HI-STORM 100 System is a radially symmetric structure, the two horizontal seismic accelerations can be combined vectorially and applied as an overturning force at the C.G. of the cask. The net overturning static moment is

$$WG_H H$$

where  $W$  is the total system weight and  $G_H$  is the resultant zero period acceleration seismic loading (vectorial sum of two orthogonal seismic loads) so that  $WG_H$  is the inertia load due to the resultant horizontal acceleration. The overturning moment is balanced by a vertical reaction force, acting at the outermost contact patch radial location  $r = D_{BASE}/2$ . The resistive moment is minimized when the vertical zero period acceleration  $G_V$  tends to reduce the apparent weight of the cask. At that instant, the moment that resists "incipient tipping" is:

$$W(1 - G_V) r$$

Performing a static moment balance and eliminating  $W$  results in the following inequality to ensure a "no-overturning condition":

$$G_H + \frac{r}{H} G_V \leq \frac{r}{H}$$



Using the values of  $r$  and  $H$  for the HI-STORM 100 ( $r = 66.25"$ ,  $H = 119.20"$ ), representative combinations of  $G_H$  and  $G_V$  that satisfy the limiting equality relation are computed and tabulated below:

<b>Acceptable Net Horizontal G-Level (HI-STORM100), <math>G_H</math></b>	<b>Acceptable Vertical G-Level, <math>G_V</math></b>
0.467	0.16
0.445	0.20
0.417	0.25
0.357	0.357

We repeat the above computations using the weight and c.g. location of the HI-STORM 100S(232). Because of the lowered center of gravity positions, the maximum net horizontal “G” levels are slightly increased.

Performing the calculations for all of the MPCs gives the following results:

$H_{cg}$  (inches)

MPC-24 with storage overpack	113.89
MPC-68 with storage overpack	114.07
MPC-32 with storage overpack	114.11
MPC-24E with storage overpack	114.01

Using the values of  $r$  and  $H$  for the HI-STORM 100S(232) ( $r = 66.25"$ ,  $H = 114.11"$ ), representative combinations of  $G_H$  and  $G_V$  that satisfy the limiting equality relation are computed and tabulated below:

<b>Acceptable Net Horizontal G-Level (HI-STORM 100S(232)), <math>G_H</math></b>	<b>Acceptable Vertical G-Level, <math>G_V</math></b>
0.488	0.16
0.464	0.20
0.435	0.25
0.367	0.367

The limiting values of  $G_H$  and  $G_V$  for the HI-STORM 100S(243), which is taller than the HI-STORM 100S(232), are the same as the HI-STORM 100.

## Primary Stresses in the HI-STORM 100 Structure Under Net Lateral Load Over 180 degrees of the Periphery

Under a lateral loading, the storage overpack will experience axial primary membrane stress in the inner and outer shells as it resists bending as a “beam-like” structure. Under the same kind of lateral loading over one-half of the periphery of the cylinder, the shells will tend to ovalize under the loading and develop circumferential stress. Calculations for stresses in both the axial and circumferential direction are required to demonstrate satisfaction of the Level D structural integrity requirements and to provide confidence that the MPC will be readily removable after a seismic event, if necessary. An assessment of the stress state in the structure under the seismic induced load will be shown to bound the results for any other condition that induces a peripheral load around part of the HI-STORM 100 storage overpack perimeter. The specific analyses are performed using the geometry and loading for the HI-STORM 100; the results obtained for stress levels and the safety assessment are also applicable to an assessment of the HI-STORM 100S.

A simplified calculation to assess the flexural bending stress in the HI-STORM 100 structure under the limiting seismic event (at which tipping is incipient) is presented in the following:

From the acceptable acceleration table presented above, maximum horizontal acceleration is bounded by 0.47g. The corresponding lateral seismic load, F, is given by

$$F = 0.47 W$$

This load will be maximized if the upper bound HI-STORM 100 weight ( $W = 360,000$  lbs. (Table 3.2.1)) is used. Accordingly,

$$F = (0.47) (360,000) = 169,200 \text{ lbs.}$$

No dynamic amplification is assumed as the overpack, considered as a beam, has a natural frequency well into the rigid range.

The moment, M, at the base of the HI-STORM 100 due to this lateral force is given by

$$M = \frac{F H}{2}$$

where H = height of HI-STORM 100 (taken conservatively as 235 inches). Note that the loading has now been approximated as a uniform load acting over the full height of the cask.

The flexural stress,  $\sigma$ , is given by the ratio of the moment M to the section modulus of the steel shell structure, z, which is computed to be 12,640 in<sup>3</sup> (Structural Calculation Package HI-981928).

Therefore,

$$\sigma = \frac{(169,200)(235)}{(12,640)(2)} = 1,573 \text{ psi}$$

We note that the strength of concrete has been neglected in the above calculation.

The maximum axial stress in the storage overpack shell will occur on the "compressive" side where the flexural bending stress algebraically sums with the direct compression stress  $\sigma_d$  from vertical compression.

From the representative acceleration table the vertical seismic accelerations corresponding to the net 0.47g horizontal acceleration is below 0.16g.

Therefore, using the maximum storage overpack weight (bounded by 270,000 lbs. from data in Table 3.2.1)

$$\sigma_d = \frac{(270,000)(1.16)}{554.47} = 565 \text{ psi}$$

where 554.47 sq. inch is the metal area (cross section) of the steel structure in the HI-STORM 100 storage overpack as computed in Subsection 3.4.4.3.2.1. The total axial stress, therefore, is

$$\sigma_T = 1,573 + 565 = 2,138 \text{ psi}$$

Per Table 3.1.12, the allowable membrane stress intensity for a Level D event is 39,750 psi at 350 degrees F.

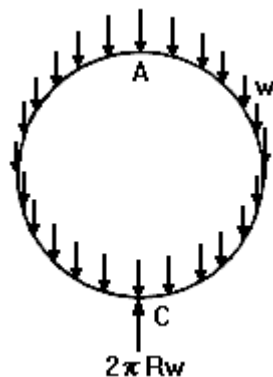
The Factor of Safety,  $\beta$ , is, therefore

$$\beta = \frac{39,750}{2,138} = 18.59$$

Examination of the *stability calculations for the overpack outer shell under a 45-g vertical end drop* results for the stability load case 2 (which considers bounding loads) in Appendix 3.AK demonstrates that no instability will result from this compressive load induced by a seismic or other environmental load leading to bending of the storage overpack as a beam.

The previous calculation has focussed on the axial stress in the members developed assuming that the storage overpack does not overturn but resists the lateral load by remaining in contact with the ground and bending like a beam. Since the lateral loading is only over a portion of the periphery, there is also the potential for this load to develop circumferential stress in the inner and outer shells to resist ovalization of the shells. To demonstrate continued retrievability of the MPC after a seismic event, it must be shown that either the stresses remain in the elastic range or that any permanent deformation that develops due to plasticity does not intrude into the MPC envelope after the event is ended. In the following subsection, *classical formulas* ~~a classical result from Appendix 3.B~~ for the deformation of rings under specified surface loadings *are* ~~is~~ used to provide a conservative solution for the circumferential stresses. Specifically, ~~the Appendix 3.B contains a complete~~ solution for a point-supported ring subject to a gravitational induced load, *as depicted in the sketch below, is implemented* ~~around the periphery of the ring~~. This solution provides a conservative estimate of the circumferential stress and the deformation of the ring that will develop under the actual applied seismic load. ~~Specifically, the following classical ring problem, shown in the sketch below, is applied to obtain the circumferential stress and deformation field under the postulated seismic event:~~

Ring supported at base and loaded by its own weight,  $w$ , given per unit circumferential length.



The solution ~~in Appendix 3.B~~ considers the geometry and load appropriate to a unit length of the inner and outer shells of the HI-STORM 100 storage overpack with a total weight equal to the overpack bounding weight (no MPC) subject to a 45g deceleration inertial loading. The numerical results *for the 45g tipover event* ~~in Appendix 3.B~~ can be directly applied here by multiplying by the factor “X”, where “X” reflects the differences in the decelerations and the weights used for the *tipover event* ~~case considered in Appendix 3.B~~ and for the seismic load case here in this subsection.

$$X = (0.47g/45g) \times (360,000\text{lb.}/270,000\text{lb.}) = 0.0139$$

Using this factor on the *tipover* solution in Appendix 3-B, (~~Attachment B-1, Case 15.16~~) gives the following bounding results for maximum stresses (without regard for sign and location of the stress) and deformations:

Maximum circumferential stress due to bending moment = (29,310 psi x X) = 407 psi

Maximum circumferential stress due to mean tangential force = (18,900 lb./2 sq.inch) x X = 131.4 psi

Change in diameter in the direction of the load = -0.11" x X = -0.0015"

Change in diameter perpendicular to the direction of the load = +0.06" x X = 0.0008"

From the above results, it is clear that no permanent ovalization of the storage overpack occurs during the seismic event and that circumferential stresses will remain elastic and are bounded by the stresses computed based on considering the storage overpack as a simple beam. Therefore, the safety factors based on maximum values of axial stress are appropriate. The magnitudes of the diameter changes that are suggested by the ring solution clearly demonstrate that ready retrievability of the MPC is maintained after the seismic event.

Because of the low values for the calculated axial stress, the conclusions of the previous section are also valid for the HI-STORM 100S.

### Potential for Concrete Cracking

It can be readily shown that the concrete shielding material contained within the HI-STORM 100 structure will not crack due to the flexuring action of HI-STORM 100 during a bounding seismic event that leads to a maximum axial stress in the storage overpack. For this purpose, the maximum axial strain in the steel shell is computed by dividing the tensile stress developed by the seismic G forces (for the HI-STORM 100, for example) by the Young's Modulus of steel.

$$\zeta = \frac{1,321}{28E+06} = 47.E-06$$

where the Young's Modulus of steel is taken from Table 3.3.2 at 350 degrees F.

The acceptable concrete strain in tension is estimated from information in ACI-318.1 for plain concrete. The ratio of allowable tensile stress to concrete Young' Modulus is computed as

$$\text{Allowable Concrete Strain} = (5 \times (0.75) \times (f)^{1/2}) / (57,000(f)^{1/2}) = 65.8E-06$$

In the above expression, f is the concrete compressive strength.

Therefore, we conclude that considerable margins against tensile cracking of concrete under the bounding seismic event exist.

## Sliding Analysis

An assessment of sliding of the HI-STORM 100 System on the ISFSI pad during a postulated limiting seismic event is performed using a one-dimensional "slider block on friction supported surface" dynamic model. The results for the shorter HI-STORM 100S are comparable. The HI-STORM 100 is simulated as a rigid block of mass 'm' placed on a surface which is subject to a sinusoidal acceleration of amplitude 'a'. The coefficient of friction of the block is assumed to be reduced by a factor  $\alpha$  to recognize the contribution of vertical acceleration in the most adverse manner (vertical acceleration acts to reduce the downward force on the friction interface). The equation of motion for such a "slider block" is given by:

$$m\ddot{x} = R + m a \sin \omega t$$

where:

- $\ddot{x}$ : relative acceleration of the slider block (double dot denotes second derivative of displacement 'x' in time)
- a: amplitude of the sinusoidal acceleration input
- $\omega$ : frequency of the seismic input motion (radians/sec)
- t: time coordinate

R is the resistive Coulomb friction force that can reach a maximum value of  $\mu(mg)$  ( $\mu$ = coefficient of friction) and which always acts in the direction of opposite to  $\dot{x}(t)$ .

Solution of the above equation can be obtained by standard numerical integration for specified values of m, a,  $\omega$  and  $\alpha$ . The following input values are used.

$$a = 0.47g$$

$$\alpha = 0.84 = 1 - \text{vertical acceleration (vertical acceleration is } 0.16g \text{ for net horizontal acceleration equal to } 0.47 \text{ from the acceleration table provided in the foregoing)}$$

$$m = 360,000 \text{ lbs/g}$$

$$\mu = 0.25$$

For establishing the appropriate value of  $\omega$ , reference is made to the USAEC publication TID-7024, "Nuclear Reactor and Earthquakes", page 35, 1963, which states that the significant energy of all seismic events in the U.S. essentially lies in the range of 0.4 to 10 Hz. Taking the mid-point value

$$\omega = (6.28) (0.5) (0.4+10) = 32.7 \text{ rad/sec.}$$

The numerical solution of the above equation yields the maximum excursion of the slider block  $x_{\max}$  as 0.12 inches, which is negligible compared to the spacing between casks.

Calculations performed at lower values of  $\omega$  show an increase in  $x_{\max}$  with reducing  $\omega$ . At 1 Hz, for example,  $x_{\max} = 3.2$  inches. It is apparent from the above that there is a large margin of safety against inter-module collision within the HI-STORM 100 arrays at an ISFSI, where the minimum installed spacing is over 2 feet (Table 1.4.1).

The above dynamic analysis indicates that the HI-STORM 100 System undergoes minimal lateral vibration under a seismic input with net horizontal ZPA g-values as high as 0.47 even under a bounding (from below) low interface surface friction coefficient of 0.25. Data reported in the literature (ACI-349R (97), Commentary on Appendix B) indicates that values of the coefficient of friction,  $\mu$ , as high as 0.7 are obtained at steel/concrete interfaces.

To ensure against unreasonably low coefficients of friction, the ISFSI pad design may require a “broom finish” at the user’s discretion. The bottom surface of the HI-STORM 100 is manufactured from plate stock (i.e. non-machine finish). A coefficient of friction value of 0.53 is considered to be a conservative numerical value for the purpose of ascertaining the potential for incipient sliding of the HI-STORM 100 System. The coefficient of friction is required to be verified by test (see Table 2.2.9).

The relationship between the vertical ZPA,  $G_V$ , (conservatively assumed to act opposite to the normal gravitational acceleration), and the resultant horizontal ZPA  $G_H$  to insure against incipient sliding is given from static equilibrium considerations as:

$$G_H + \mu G_V \leq \mu$$

Using a conservative value of  $\mu$  equal to 0.53, the above relationship provides governing ZPA limits for a HI-STORM 100 (or 100S) System arrayed in a freestanding configuration. The table below gives representative combinations that meet the above limit.

$G_H$ (in g's)	$G_V$ (in g's)
0.445	0.16
0.424	0.20
0.397	0.25
0.350	0.34

If the values for the DBE event at an ISFSI site satisfy the above inequality relationship for incipient sliding with coefficient of friction equal to 0.53, then the non-sliding criterion set forth in NUREG-1536 is assumed to be satisfied a priori. However, if the ZPA values violate the inequality by a small amount, then it is permissible to satisfy the non-sliding criterion by implementing measures to roughen the HI-STORM 100/ISFSI pad interface to elevate the value of  $\mu$  to be used in the inequality relation. To demonstrate that the value of  $\mu$  for the ISFSI pad meets the required value implied by the above inequality, a series of Coulomb friction (under the QA program described in Chapter 13) shall be performed as follows:

Pour a concrete block with horizontal dimensions no less than 2' x 2' and a block thickness no less than 0.5'. Finish the top surface of the block in the same manner as the ISFSI pad surface will be prepared.

Prepare a 6" x 6" x 2" SA516 Grade 70 plate specimen (approximate weight = 20.25 lb.) to simulate the bottom plate of the HI-STORM 100 overpack. Using a calibrated friction gage attached to the steel plate, perform a minimum of twenty (20) pull tests to measure the static coefficient of friction at the interface between the concrete block and the steel plate. The pull tests shall be performed on at least ten (10) different locations on the block using varying orientations for the pull direction.

The coefficient of friction to be used in the above sliding inequality relationship will be set as the average of the results from the twenty tests.

The satisfaction of the "no-sliding" criterion set down in the foregoing shall be carried out along with the "no-overturning" qualification (using the static moment balance method in the manner described at the beginning of this subsection) and documented as part of the ISFSI facility's CFR72.212 evaluation.

#### Alternative Evaluation of Overturning and Sliding

*In this subsection, an evaluation of the propensity for the free standing cask to be in a state of either incipient overturning or incipient sliding has been performed using a simple static analysis that is independent of time phasing of the input acceleration time histories and considers only the Zero Period Acceleration (ZPA) obtained from the response spectra. For both incipient overturning and incipient sliding, the following inequality must be satisfied to ensure satisfaction of the static criteria.*

$$G_H + \mu G_V \leq \mu$$

*For the incipient overturning evaluation,  $\mu$ =(radius of cask base/height to loaded cask center-of-gravity). For the incipient sliding evaluation,  $\mu$ =Coulomb coefficient of friction =0.53 at the cask/ISFSI pad interface (unless testing justifies use of a higher value). The inequality has been derived assuming that the cask is resting on a flat and level surface that is subject to a seismic event characterized by a response spectra set with the net horizontal and vertical Zero Period Acceleration (ZPA) denoted by  $G_H$  and  $G_V$ , respectively.*

*This "screening" evaluation provides a conservative criterion to insure that top-of-pad acceleration time histories from the aggregate effect of soil structure interaction and free field acceleration would not predict initiation of overturning or sliding. If on-the-pad acceleration time histories are available, the applicable inequality (for overturning and sliding) may be satisfied at each time instant during the Design Basis Earthquake with  $G_H$  and  $G_V$  representing coincident values of the magnitude of the net horizontal and vertical acceleration vectors.*



### 3.4.7.2 Explosion (Load Case 05 in Table 3.1.5)

In the preceding subsection, it has been demonstrated that incipient tipping of the storage overpack will not occur under a side load equal to 0.47 times the weight of the cask. For a fully loaded cask, this side load is equal to  
 $F = 169,200 \text{ lb.}$

If it is assumed that this side load is uniformly distributed over the height of the cask and that the cask centroid is approximately at the half-height of the overpack, then an equivalent pressure,  $P$ , acting over 180 degrees of storage overpack periphery, can be defined as follows:

$$P \times (DH) = F$$

Where  $D$  = overpack outside diameter, and  $H$  = height of storage overpack

For  $D = 132.5''$  and  $H = 235''$ , the equivalent pressure is

$$P = 169,200 \text{ lb}/(132.5'' \times 235'') = 5.43 \text{ psi}$$

Therefore, establishing 5 psi as the design basis steady state pressure differential (Table 2.2.1) across the overpack diameter ensures that incipient tipping will not occur.

Since the actual explosion produces a transient wave, the use of a static incipient tip calculation is very conservative. To evaluate the margin against tip-over from a short-time pressure pulse, a Working Model analysis of the two-dimensional dynamic motion of the HI-STORM subject to a given initial angular velocity is carried out. Figures 3.4.25 and 3.4.26 provide details of the model and the solution for a HI-STORM 100 System (simulated as a rigid body) having a weight and inertia property appropriate to a minimum weight cask. The results show that an initial angular velocity of 0.626 radians/second does not lead to a tipover of the storage overpack. The results bound those obtained for the HI-STORM 100S(232) since the overall cask height is reduced. The results for the HI-STORM 100S(243) are roughly equal to the results for the HI-STORM 100 since the differences in height and weight are negligible.

The initial angular velocity can be related to a square wave pressure pulse of magnitude  $P$  and time duration  $T$  by the following formula:

$$I\omega = (P \times D \times H) \times (0.5 \times H) \times T$$

The above formula relates the change in angular motion resulting from an impulsive moment about the base of the overpack.  $D$  is the diameter of the outer shell,  $H$  is the height of the storage overpack, and  $I$  is the mass moment of inertia of the storage overpack about the mass center (assumed to be at half-height). For  $D=132.5''$ ,  $H=235''$ ,  $P=10 \text{ psi}$ ,  $T=1 \text{ second}$ , and  $I=64,277,000 \text{ lb.inch sec}^2$  (calculated in Appendix 3.C), the resulting initial angular velocity is:

$$\omega = 0.569 \text{ radians/second}$$



$$P \times D = 56,180 \text{ lb./inch} \quad D = 132.5''$$

$$P = 424 \text{ psi}$$

Since this is higher by a large margin than any postulated external pressure load, circumferential stresses induced by the differential pressure specified in Table 2.2.1 are insignificant. Specifically, by adding the results from the two solutions (ring load case 1 for a point support reaction to a body force + ring load case 3 for a point support reaction to a lateral pressure over one-half of the perimeter) ~~considered in Appendix 3.B~~, it is determined that the circumferential bending stress from case 1 ~~in that appendix~~ is reduced by the factor “R” to obtain the corresponding stress from the combined case. R is computed as the ratio of moment magnitudes from the combined case to the results of case 1 alone.

$$R = (\text{maximum bending moment from case 1 + case 3})/(\text{maximum bending moment from case 1}) \\ = 0.75/6.197 = 0.12$$

~~(results for individual cases are in Appendix 3.B)~~

Examination of the graphs from the moment distribution from the two solutions ~~in Appendix 3.B~~ shows that the individual terms always subtract and nearly cancel each other at every location.

Therefore, it is concluded that the maximum circumferential stress that develops under a pressure of 424 psi applied over one-half of the perimeter, and conservatively assumed balanced by inertia loading, is

$$\text{Stress} = 29,310 \text{ psi} \times 0.12 = 3517 \text{ psi}$$

The stress due to a differential pressure of 10 psi (Table 2.2.1) is only 2.36% of the above value and needs no further evaluation for stress limits or deformation to demonstrate retrievability of the MPC.

#### 3.4.7.3 Anchored HI-STORM Systems Under High-Seismic DBE (Load Case C in Table 3.1.1)

The anchored HI-STORM System (Figures 1.1.4 and 1.1.5) is assumed to be subjected to quasi-static inertial seismic loads corresponding to the ZPA design basis limits given in Table 2.2.8. The results from this quasi-static analysis are used to evaluate structural margins for the preloaded anchor studs and the sector lugs. In the quasi-static evaluation, the effect of the “rattling” of the MPC inside of the overpack is accounted for by the imposition of a dynamic load factor of 2.0 on the incremental stresses that arise during the seismic event. In addition to the quasi-static analysis, confirmatory 3-D dynamic analyses are performed using base acceleration excitation histories developed from two sets of response spectra. Figure 3.4.30 shows the two sets of response spectra that are assumed to be imposed at the top of the ISFSI pad. One set of response spectra is the Regulatory Guide 1.60 spectra for 5% damping with zero period acceleration conservatively amplified to 1.5 in each direction. This spectra set has been used as the input spectra at many nuclear plants in the U.S. (although generally, the ZPA was much below 1.0). Three statistically independent

acceleration time histories (two horizontal labeled as “H1”, “H2”) and one vertical (labeled as “VT”) have been developed. A twenty-second duration event was considered. Figures 3.4.31 to 3.4.33 show the time histories. The second set of response spectra used for time history analysis has similar levels of zero period acceleration but has higher peak spectral acceleration values in the low frequency range (2-3 Hz). This spectra set is the design basis set for a Pacific coast U.S. plant. Figures 3.4.34 to 3.4.36 ( labeled as “FN”, “FP” for the two horizontal acceleration histories and “FV” for the vertical acceleration time history), show the corresponding time histories simulating a long duration seismic event (170 seconds).

The objectives of the quasi-static and dynamic seismic analyses are the following:

- i. Quantify the structural safety factor in the anchor studs and in the sector lugs that constitute the fastening system for the loaded HI-STORM 100A overpack. The structural safety factor is defined as the ratio of the permitted stress (stress intensity) per Subsection “NF” of the ASME Code to the maximum stress (stress intensity) developed in the loaded component.
- ii. Compute the safety factor against fatigue failure of the anchor studs from a single seismic event.
- iii. Quantify the interface loads applicable to the ISFSI pad to enable the ISFSI owner to design the ISFSI pad under the provisions of ACI-349 (85). The bounding interface loads computed for the maximum intensity seismic event (ZPA) and for extreme environmental loads may be used in pad design instead of the site-specific loads calculated for the loadings applicable to the particular ISFSI.

The above design objectives are satisfied by performing analyses of a loaded HI-STORM 100A System using a conservative set of input data and a conservative dynamic model. Calculations using the quasi-static model assume that the net horizontal inertia loads and the vertical inertia load correspond to the weight of the loaded cask times the appropriate ZPA. The results from the analyses are set down as the interface loads, and may be used in the ISFSI pad design work effort by the ISFSI owner. The information on the seismic analysis is presented in five paragraphs as follows:

- Input data for analysis
- Quasi-static model and results
- Dynamic model and modeling assumptions.
- Results of dynamic analysis
- Summary of interface loads

a. Input Data for Analysis:

Key input data for the seismic analysis of a loaded HI-STORM 100A System is summarized in Table 3.4.10. As can be seen from Table 3.4.10, the input data used in the analysis is selected to bound the actual data, wherever possible, so as to maximize the seismic response. For example, a bounding weight of the loaded MPC and HI-STORM 100A overpack is used because an increase in the weight of the system directly translates into an increased inertial loading on the structure.

For quasi-static analysis, bounding ZPA values of 1.5 in all three directions are used with the vertical event directed upward to maximize the stud tension. The resulting ZPA's are then further amplified by the dynamic load factor (DLF=2.0) to reflect "rattling" of the MPC within the overpack. Input data for anchor stud lengths are representative. We consider long and short studs in order to evaluate the effect of stud spring rate.

For the confirmatory dynamic analyses, the time history base excitations are shown in Figures 3.4.31 through 3.4.36 and the propensity for "rattling" is included in the model.

b. Quasi-Static Model and Results:

We consider the HI-STORM100A baseplate as a rigid plate resting on the ISFSI pad with the twenty-eight studs initially preloaded so as to impart a compressive load at the baseplate pad interface that is balanced by a tensile load in the studs prior to the seismic event occurring. The discrete studs are replaced by a thin ring located at the stud circle radius for analysis purposes. The thickness of the thin ring is set so that the ring area is equal to the total stress area of the twenty-eight studs. Figure 3.4.37 shows a view of a segment of the baseplate with the outline of the ring. The ISFSI pad is represented by a linear spring and a rotational spring with spring constants determined from the exact solution for a rigid circular punch pressed into a elastic half-space. We assume that subsequent to pre-tensioning the studs, the seismic event occurs, represented by a net horizontal load DH and a net vertical load DV. In the analysis, the input loads DH and DV are:

$$G_H = (1.5^2 \times 2)^{1/2} \times DLF = 4.242 ; \quad G_V = 1.5 \times DLF = 3.0$$

$$DH = G_H \times 360,000 \text{ lb.} ; \quad DV = -G_V \times 360,000 \text{ lb}$$

DH is the magnitude of the vector sum of the two horizontal ZPA accelerations multiplied by the bounding HI-STORM 100A weight. Similarly, DV is an upward directed load due to the vertical ZPA acceleration. The upward direction is chosen in order to maximize the stud tension as the assemblage of studs and foundation resists overturning from the moment induced by DH applied at the centroid of the cask. Figure 3.4.38 shows the free-body diagram associated with the seismic event. Essentially, we consider an analysis of a pre-compressed interface and determine the interface joint behavior under the imposition of an external loading (note that this kind of analysis is well established in the pressure vessel and piping area where it is usually associated with establishing the effectiveness of a gasketed joint). An analysis is performed to determine the maximum stud tension that results if the requirement of no separation between baseplate and pad is imposed under the imposed loading. The following result is obtained from static equilibrium, for a preload stress of 60 ksi, when the "no separation condition" is imposed:

$$\frac{2a/3h_{cg} (F_{\text{preload}}/W + 1)(1 + \alpha_1)}{G_H - 2a/3h_{cg} (G_V (1 + \alpha_1)/(1 + \alpha))} = 1.016$$

In the above equation,

$$F_{\text{preload}} = (\text{Total stress area of twenty-eight, 2" diameter studs}) \times 60 \text{ ksi} = 4,200,000 \text{ lb.}$$

$$W = \text{Bounding weight of loaded HI-STORM 100A} = 360,000 \text{ lb.}$$

$$a = 73.25 \text{ inches,}$$

$$h_{\text{cg}} = 118.5 \text{ inches}$$

The coefficients  $\alpha$  and  $\alpha_1$  relate the stiffness of the totality of studs to the stiffness of the foundation under direct loading and under rotation. The result given above is for the representative case of stud free length "L", equal to

L= 42 inches, which gives  $\alpha$  and  $\alpha_1$  equal to 0.089 and 0.060, respectively.

A simplified confirmatory analysis of the above problem can be performed by considering the limiting case of a rigid baseplate and a rigid ISFSI pad. In the limit of a rigid ISFSI pad (foundation), the coefficients  $\alpha$  and  $\alpha_1$  go to zero. A related solution for the case of a rigid baseplate and a rigid foundation can be obtained when the criteria is not incipient separation, but rather, a more "liberal" incipient rotation about a point on the edge of the baseplate. That solution is given in "Mechanical Design of Heat Exchangers and Pressure Vessel Components", by Singh and Soler (Arcturus Publishers, 1984). The result is (for 60 ksi prestress in each stud):

$$\frac{a/h_{\text{cg}} (F_{\text{preload}}/W + 1)}{G_H - a/h_{\text{cg}} (G_V)} = 1.284$$

Although not a requirement of any design code imposed herein, the right hand side of the previous relationships can be viewed as the safety factor against incipient separation (or rotation about an edge) at the radius "a". Note that since we have assumed a bounding event, there is an additional margin of 1.5 in results since the Reg. Guide 1.60 event has not been applied with a ZPA in excess of 1.0.

For the real seismic event associated with a western U.S. plant having a slightly lower horizontal ZPA and a reduced vertical ZPA (see Figure 3.4.30). Using the same DLF =2.0 to account for "rattling" of the confined MPC:

$$G_H = 4.1 \quad ; \quad G_V = 2.6,$$

the aforementioned safety factors are:

$$\text{SF (incipient separation)} = 1.076$$

$$\text{SF (incipient edging)} = 1.372$$

The increment of baseplate displacement and rotation, up to incipient separation, is computed from the equilibrium and compatibility equations associated with the free body in Figure 3.4.38 and the change in stud tension computed. The following formula gives the stud tensile stress in terms of the initial preload and the incremental change from the application of the horizontal and vertical seismic load.

$$\sigma_{\text{stud}} = \sigma_{\text{preload}} + \alpha \frac{W}{NA_{\text{stress}}} \left( \frac{-G_V}{1 + \alpha} + \left( \frac{3h_{\text{cg}}}{2a} \right) \left( \frac{c}{a} \right) \left( \frac{G_H}{1 + \alpha_1} \right) \right)$$

In the above formula,

N = number of studs = 28 (maximum number based on HI-STORM dimensions). For lower seismic inputs, this might be reduced (in groups of 4 to retain symmetry).

A<sub>stress</sub> = tensile stress area of a 2” diameter stud

2c = stud circle diameter

The results demonstrate that there is a relatively small change in stud stress from the initial pre-tension condition with the ISFSI pad foundation resisting the major portion of the overturning moment. For the geometry considered (maximum stud free length and nominal prestress), the maximum tensile stress in the stud increases by 9.1%. The following table summarizes the results from the quasi-static analysis using minimum ultimate strength for the stud to compute the safety factors. Note that under the seismic load, the direct stress in the stud is limited to 70% of the stud ultimate strength (per Appendix F of the ASME Code Section III). The allowable pad compressive stress is determined from the ACI Code assuming confined concrete and the minimum concrete compressive strength from Table 2.0.4. Because of the large compressive load at the interface from the pre-tensioning operation, the large frictional resistance inhibits sliding of the cask. Consequently, there will be no significant shear stress in the studs. Safety factors for sliding are obtained by comparing the ratio of horizontal load to vertical load with the coefficient of friction between steel and concrete (0.53). Values in parenthesis represent results obtained using ZPA values associated with the real seismic event for the western U.S. plant instead of the bounding Reg. Guide 1.60 event.

<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI-STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 60 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	65.48 (65.18)	87.5	1.336 (1.343)
Maximum Pad Pressure (ksi)(42" stud free length)	3.126 (3.039)	4.76	1.52 (1.57)
Stud Stress (ksi)(16" stud free length)	73.04 (72.34)	87.5	1.20 (1.21)
Maximum Pad Pressure(ksi) (16" stud free length)	2.977 (2.898)	4.76	1.60 (1.64)
Overpack Sliding	0.439 (0.407)	0.53	1.21 (1.31)

The effect of using a minimum stud free length in the embedment design is to increase the values of the coefficients  $\alpha$  and  $\alpha_1$  because the stud stiffness increases. The increase in stud stiffness, relative to the foundation stiffness results in an increase in incremental load on the studs. This is a natural and expected characteristic of preloaded configurations. It is noted that the stud safety factors are based on minimum ultimate strength and can be increased, without altering the calculated results, by changing the stud material.

The quasi-static analysis methodology has also been employed to evaluate the effects of variation in the initial prestress on the studs. The following tables reproduce the results above for the cases of lower bound stud prestress (55 ksi) and upper bound stud prestress (65 ksi) on the studs. Only the results using the values associated with the Reg. Guide 1.60 bounding event are reported.

<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI-STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 55 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	60.48	87.5	1.45
Maximum Pad Pressure (ksi)(42" stud free length)	3.012	4.76	1.58
Stud Stress (ksi)(16" stud free length)	68.07	87.5	1.29
Maximum Pad Pressure(ksi) (16" stud free length)	2.862	4.76	1.663
Overpack Sliding	0.488	0.53	1.09



<b>SUMMARY OF RESULTS FOR STUDS AND INTERFACE FROM QUASI- STATIC SEISMIC EVALUATION WITH DLF = 2.0, Stud Prestress = 65 ksi</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Stud Stress(ksi) (42" stud free length)	70.48	87.5	1.24
Maximum Pad Pressure (ksi)(42" stud free length)	3.24	4.76	1.47
Stud Stress (ksi)(16" stud free length)	78.07	87.5	1.12
Maximum Pad Pressure(ksi) (16" stud free length)	3.091	4.76	1.54
Overpack Sliding	0.399	0.53	1.33

The results above confirm the expectations that an increase in preload increases the safety factor against sliding. The calculated coefficient of friction in the above tables is computed as the ratio of applied horizontal load divided by available vertical load. For all combinations examined, ample margin against incipient separation at the interface exists.

Based on the results from the quasi-static analysis, an assessment of the safety factors in the sector lugs is obtained by performing a finite element analysis of a repeated element of one of the sector lugs. Figure 3.4.39 shows the modeled section and the finite element mesh. The stud load is conservatively applied as a uniform downward pressure applied over a 5"x5" section of the extended baseplate simulating the washer between two gussets. This is conservative as the rigidity of the washer is neglected. The opposing pressure loading from the interface pressure is applied as a pressure over the entire extended baseplate flat plate surface. Only one half the thickness of each gusset plate is included in the model. Two cases are considered: (1) the pre-loaded state (a Normal Condition of Storage-Level A stress limits apply); and, (2), the seismic load condition at the location of the maximum tensile load in a stud (an Accident Condition of Storage – Level D stress intensity limits apply). Figures 3.4.40 and 3.4.41 present the stress results for the following representative input conditions:

Level A analysis - Preload stress/bolt = 60 ksi

Level D analysis - Maximum Bolt stress(includes seismic increment) = 65.5 ksi

In the Level A analysis, the resisting local foundation pressure exactly balances the preload. For the Level D analysis, the opposing local foundation pressure = 190 psi (average over the area between gussets. This represents the reduced pressure under the highest loaded stud under the induced rotation of the storage system.

The most limiting weld stress is obtained by evaluating the available load capacity of the fillet weld attaching the extended baseplate annulus region to the gussets (approximately 25 inches of weld per segment) using a limit strength equal to 42% of the ultimate strength of the base material.

The following table summarizes the limiting safety factors for the sector lugs. Allowable values for primary bending stress and stress intensity are from Tables 3.1.10 and 3.1.12 for SA-516 Grade 70 @ 300 degrees F.

<b>SUMMARY OF RESULTS FOR SECTOR LUGS FROM QUASI-STATIC SEISMIC EVALUATION</b>			
<b>Item</b>	<b>Calculated Value</b>	<b>Allowable Value</b>	<b>Safety Factor = (Allowable Value/Calculated Value)</b>
Maximum Primary Membrane + Bending Stress Away From Loaded Region and Discontinuity (ksi) – Case 1 - Preload	15.62	26.3	1.68
Maximum Primary Membrane + Bending Stress Intensity Away From Loaded Region and Discontinuity (ksi) – Case 2 - Preload + Seismic	36.67	60.6	1.65
Maximum Weld Shear Load (kips)	150.8	194.9	1.29

c. Dynamic Model and Modeling Assumptions:

The dynamic model of the HI-STORM 100A System consists of the following major components.

- i. The HI-STORM 100 overpack is modeled as a six degree-of-freedom (rigid body) component.
- ii. The loaded MPC is also modeled as a six degree-of-freedom (rigid body) component that is free to rattle inside the overpack shell. Gaps between the two bodies reflect the nominal dimensions from the drawings.
- iii. The contact between the MPC and the overpack is characterized by a coefficient of restitution and a coefficient of friction. For the dynamic analysis, the coefficient of restitution is set to 0.0, reflecting the large areas of nearly flat surface that come into contact and have minimal relative rebound. The coefficient of friction is set to 0.5 between all potentially contacting surfaces of the MPC/overpack interface.
- iv. The anchor studs, preloaded to axial stress  $\sigma_i$  (Table 3.4.10), induce a contact stress between the overpack base and the ISFSI pad. The loaded cask-pad interface can support a certain amount of overturning moment before an uplift (loss of circularity of the contact patch) occurs. The anchor studs are modeled as individual linear springs connecting the periphery of the extended baseplate to the ISFSI pad section. The resistance of the foundation is modeled by a vertical linear spring and three rotational springs connected between the cask baseplate center point and the surface of the flat plate modeling the driven ISFSI pad. The ISFSI pad is driven with the three components of acceleration time history applied simultaneously.

The HI-STORM 100A dynamic model described above is implemented on the public domain computer code WORKING MODEL (also known as VisualNastran) (See Subsection 3.6.2 for a description of the algorithm).

Figures 3.4.42 and 3.4.43 show the rigid body components of the dynamic model before and after assembly. The linear springs are not shown. Mass and inertia properties of the rigid bodies are consistent with the bounding property values in Table 3.4.10.

*e.d.* Results of Dynamic Analysis:

Figures 3.4.44–3.4.47 show results of the dynamic analysis using the Reg. Guide 1.60 seismic time histories as input accelerations to the ISFSI pad. Figure 3.4.44 shows variation in the vertical foundation compressive force. Figure 3.4.45 shows the corresponding load variation over time for the stud having the largest instantaneous tensile load. An initial preload of approximately 150,000 lb is applied to each stud (corresponding to 60,160 psi stud tensile stress). This induces an initial compression load at the interface approximately equal to 571,000 lb. (including the dead weight of the loaded HI-STORM). Figures 3.4.44 and 3.4.45 clearly demonstrate that the foundation resists the majority of the oscillatory and impactive loading as would be expected of a preloaded configuration. Figure 3.4.46 shows the impulse (between the MPC and HI-STORM 100A) as a function of time. It is clear that the “spikes” in both the foundation reaction and the stud load over the total time of the event are related to the impacts of the rattling MPC. The results provide a graphic demonstration that the rattling of the MPC inside the overpack must be accounted for in any quasi-static representation of the event. The quasi-static results presented herein for the anchored system, using a DLF = 2.0, are in excellent agreement with the dynamic simulation results.

We note that the dynamic simulation, which uses an impulse-momentum relationship to simulate the rattling contact, leads to results having a number of sharp peaks. Given that the stress intensity limits in the Code assume static analyses, filtering of the dynamic results is certainly appropriate prior to comparing with any static allowable strength. We conservatively do not perform any filtering of the results prior to comparison with the quasi-static analysis; we note only that any filtering of the dynamic results to eliminate high-frequency effects resulting from the impulse-momentum contact model would increase the safety factors. Finally, Figure 3.4.47 shows the ratio of the net interface horizontal force (needed to maintain equilibrium) to the instantaneous compression force at the ISFSI pad interface with the base of the HI-STORM 100A. This ratio, calculated at each instant of time from the dynamic analysis results using the Reg. Guide 1.60 event, represents an instantaneous coefficient of friction that is required to ensure no interface relative movement. Figure 3.4.47 demonstrates that the required coefficient of friction is below the available value 0.53. Thus, the dynamic analysis confirms that the foundation interface compression, induced by the preloading action, is sufficient to maintain a positive margin against sliding without recourse to any resistance from the studs.

The results of the dynamic analysis using acceleration time histories from the Reg. Guide 1.60 response spectra (grounded at 1.5 g's) confirm the ability of the quasi-static solution, coupled with a dynamic load factor, to correctly establish structural safety factors for the anchored cask. The dynamic analysis confirms that stud stress excursions from the preload value are minimal despite the large overturning moments that need to be balanced.

A second dynamic simulation has been performed using the seismic time histories appropriate to a pacific coast U.S. nuclear plant (Figures 3.4.34-3.4.36). The ZPA of these time histories are slightly less than the Reg. Guide 1.60 time histories but the period of relatively strong motion extends over a longer time duration. The results from this second simulation exhibit similar behavior as those results presented above and provide a second confirmation of the validity of the safety factors predicted by the quasi-static analysis. Reference [3.4.14] (see Subsection 3.8) provides archival information and backup calculations for the results summarized here.

Stress cycle counting using Figure 3.4.45 suggests 5 significant stress cycles per second provides a bounding number for fatigue analysis. A fatigue reduction factor of 4 is appropriate for the studs (per ASME Code rules). Therefore, a conservative analysis of fatigue for the stud is based on an alternating stress range of:

$S(\text{alt}) = .5 \times (22,300 \text{ psi}) \times 4 = 44,600 \text{ psi}$  for 5 cycles per second. The value for the stress range is obtained as the difference between the largest tensile stress excursions from the mean value as indicated in the figure.

To estimate fatigue life, we use a fatigue curve from the ASME Code for high strength steel bolting materials (Figure I.9.4 in Appendix I, ASME Code Section III Appendices) For an amplified alternating stress intensity range of 44,600 psi, Figure I.9.4 predicts cyclic life of 3,000 cycles. Therefore, the safety factor for failure of a stud by fatigue during one Reg. Guide 1.60 seismic event is conservatively evaluated as:

$$SF(\text{stud fatigue}) = 3,000/100 = 30.$$

For the long duration event, even if we make the conservative assumption of a nine-fold increase in full range stress cycles, the safety factor against fatigue failure of an anchor stud from a single seismic event is 3.33. Recognizing that the fatigue curve itself is developed from test data with a safety factor of 20 on life and 4 on stress, the results herein demonstrate that fatigue failure of the anchor stud, from a single seismic event, is not credible.

e. ~~—d.~~ Summary of Interface Loads for ISFSI Pad Design:

Bounding interface loads are set down for use by the ISFSI pad designer and are based on the validated quasi-static analysis and a dynamic load factor of 2.0:

<b>BOUNDING INTERFACE LOADS FOR ISFSI PAD STRUCTURAL/SEISMIC DESIGN</b>	
D (Cask Weight)	360 kips
D (Anchor Preload @ 65 ksi)	4,550 kips
E (Vertical Load)	1,080 kips
E (Net Horizontal Surface Shear load)	1,527.35 kips
E (Overturning Moment)	15,083 kip-ft.

3.4.8 Tornado Wind and Missile Impact (Load Case B in Table 3.1.1 and Load Case 04 in Table 3.1.5)

During a tornado event, the HI-STORM 100 System is assumed to be subjected to a constant wind force. It is also subject to impacts by postulated missiles. The maximum wind speed is specified in Table 2.2.4 and the three missiles, designated as large, intermediate, and small, are described in Table 2.2.5.

In contrast to a freestanding HI-STORM 100 System, the anchored overpack is capable of withstanding much greater lateral pressures and impulsive loads from large missiles. The quasi-static analysis result, presented in the previous subsection, can be used to determine a maximum permitted base overturning moment that will provide at least the same stud safety factors. This is accomplished by setting  $G_V = 0.0$ ,  $DLF = 1$  and finding an appropriate  $G_H$  that gives equal or better stud safety factors. The resulting value of  $G^*_H$  establishes the limit overturning moment for combined tornado missile plus wind.,  $M_L$ . ( $G^*_H \times \text{Weight} \times h_{cg}$ ) is conservatively set as the maximum permissible moment at the base of the cask due to combined action of lateral wind and tornado missile loading. Thus, if the lateral force from a tornado missile impact is  $F$  at height  $h$  and that from steady tornado wind action is a resultant force  $W$  acting at cask mid-height ( $0.5H$ ), and the two loads are acting synergistically to overturn the cask, then their magnitudes must satisfy the inequality

$$0.5WH + Fh \leq M_L$$

where the limit moment is established to ensure that the safety factors for seismic load remain bounding.

$$M_L = 18,667 \text{ kip-ft.}$$

Tornado missile impact factors should be factored into “F” prior to determining the validity of the above inequality for any specific site.

In the case of a free-standing system, the post impact response of the HI-STORM 100 System is required to assess stability. Both the HI-STORM 100 storage overpack, and the HI-TRAC transfer cask are assessed for missile penetration.

~~Appendix 3.C contains~~ The results for the post-impact response of the HI-STORM 100 storage overpack where it is demonstrated there that the combination of tornado missile plus either steady tornado wind or instantaneous tornado pressure drop causes a rotation of the HI-STORM 100 to a maximum angle of inclination less than 3 degrees from vertical. This is much less than the angle required to overturn the cask. ~~The appropriate value for the drag coefficient used in the computation of the lateral force on the storage overpack from tornado wind is justified in Appendix 3.C.~~ The results for the HI-STORM 100 are bounding since the HI-STORM 100S has a lower center of gravity when loaded.

~~Appendix 3.C computes~~ The maximum force (not including the initial pulse due to missile impact) acting on the projected area of the storage overpack *is computed* to be:

$$F = 91,920 \text{ lbs.}$$

The instantaneous impulsive force due to the missile strike is not computed here; its effect is felt as an initial angular velocity imparted to the storage overpack at time equal to zero. The net resultant force due to the simultaneous pressure drop is not an all-around distributed loading that has a net resultant, but rather is more likely to be distributed only over 180 degrees (or less) of the storage overpack periphery. The circumferential stress and deformation field will be of the same order of magnitude as that induced by a seismic loading. Since the magnitude of the force due to  $F$  is less than the magnitude of the net seismically induced force considered in Subsection 3.4.7, the storage overpack global stress analysis performed in Subsection 3.4.7 remains governing. In the next subsection, results are provided for the circumferential stress and ovalization of the portion of the storage overpack due to the bounding estimate for the impact force of the intermediate missile.

#### 3.4.8.1 HI-STORM 100 Storage Overpack

~~Appendix 3.C~~ This subsection considers the post impact behavior of the HI-STORM 100 System after impact from tornado missiles. During an impact, the system consisting of missile plus storage overpack and MPC satisfies conservation of linear and angular momentum. The large missile impact is assumed to be inelastic. This assumption conservatively transfers all of the momentum from the missile to the system. The intermediate missile and the small missile are assumed to be unyielding and hence the entire initial kinetic energy is assumed to be absorbed by motion of the cask and local yielding and denting of the storage overpack surface. It is shown that cask stability is maintained under the postulated wind and large missile loads. The conclusion is also valid for the HI-STORM 100S since its lower center of gravity inherently provides additional stability margin.

The penetration potential of the missile strikes (Load Case 04 in Table 3.1.5) is examined ~~first in Appendix 3.G~~. ~~It is shown in Appendix 3.G~~ The detailed calculations show that there will be no penetration through the concrete surrounding the inner shell of the storage overpack or penetration of the top closure plate. Therefore, there will be no impairment to the confinement boundary due to missile strikes during a tornado. Since the inner shell is not compromised by the missile strike, there will be no permanent deformation of the inner shell. Therefore, ready retrievability is assured after the missile strike. The following ~~paragraphs~~ results summarize the ~~analysis work in Appendix 3.G~~.

- a. The small missile will dent any surface it impacts, but no significant puncture force is generated. The 1" missile can enter the air ducts, but geometry prevents a direct impact with the MPC.
- b. The following table summarizes the denting and penetration analysis performed for the intermediate missile ~~in Appendix 3.G~~. Denting is used to connote a local deformation mode encompassing material beyond the impacting missile envelope, while penetration is used to connote a plug type failure mechanism involving only the target material immediately under the impacting missile.

Location	Denting (in.)	Thru-Thickness Penetration
Storage overpack outer Shell	6.87	Yes (>0.75 in.)
Radial Concrete	9.27	No (<27.25 in.)
Storage overpack Top Lid	0.4	No (<4 in.)

The primary stresses that arise due to an intermediate missile strike on the side of the storage overpack and in the center of the storage overpack top lid are also determined ~~next in Appendix 3-G.~~ The analysis of the storage lid for the HI-STORM 100 bounds that for the HI-STORM 100S; because of the additional energy absorbing material (concrete) in the direct path of a potential missile strike on the top lid of the HI-STORM 100S lid, the energy absorbing requirements of the circular plate structure are much reduced. ~~It is demonstrated there.~~ *The analysis demonstrates* that Level D stress limits are not exceeded in either the overpack outer shell or the top lid. The safety factor in the storage overpack, considered as a cantilever beam under tip load, is computed, as is the safety factor in the top lids, considered as two centrally loaded plates. The applied load, in each case, is the missile impact load. A summary of the results for axial stress in the storage overpack, ~~as obtained from Appendix 3-G,~~ is given in the table below:

<b>HI-STORM 100 MISSILE IMPACT - Global Axial Stress Results</b>			
Item	Value (ksi)	Allowable (ksi)	Safety Factor
Outer Shell – Side Strike	14.35	39.75	2.77
Top Lid - (End Strike)	44.14	<del>57.05</del> 59.65	<del>1.291</del> 1.351

To demonstrate ready retrievability of the MPC, we must show that the storage overpack suffers no permanent deformation of the inner shell that would prevent removal of the MPC after the missile strike. To demonstrate ready retrievability (for both HI-STORM 100 and for HI-STORM 100S) a conservative evaluation of the circumferential stress and deformation state due to the missile strike on the outer shell ~~is was performed.~~ ~~Appendix 3-G calculates a~~ *A conservative estimate for the 8” diameter missile impact force, “Pi”, on the side of the storage overpack is calculated as:*

$$P_i = 843,000 \text{ lb.}$$

This force is conservative in that the target overpack is assumed rigid; any elasticity serves to reduce the peak magnitude of the force and increase the duration of the impact. The use of the upper bound value is the primary reason for the high axial stresses resulting from this force. To demonstrate continued ability to retrieve the MPC subsequent to the strike, circumferential stress and

deformation that occurs locally in the ring section near the location of the missile strike are investigated.

~~Results in Appendix 3.B are presented under different ring loadings. Subsection 3.4.7 presents stress and displacement results for a composite ring of unit width consisting of the inner and outer shells of the storage overpack. The solutions in Appendix 3.B assumes that the net loading is 56,184 lb. applied on the 1" wide ring (equivalent to a 45G deceleration applied uniformly along the height on a storage overpack weight of 270,000 lb.). The This solution for case 1 in Appendix 3.B can be applied directly to evaluate the circumferential stress and deformation caused by a tornado missile strike on the outer shell. Using the results for the 45g tipover event in Appendix 3.B, an attenuation factor to adjust the results from case 1 in Appendix 3.B is developed that reflects the difference in load magnitude and the width of the ring that is effective in resisting the missile strike force. The strike force  $P_i$  is resisted by a combination of inertia force and shear resistance from the portion of the storage overpack above and below the location of the strike. The ring theory solution to determine the circumferential stress and deformation conservatively assumes that inertia alone, acting on an effective length of ring, balances the applied point load  $P_i$ . The effective width of ring that balances the impact load is conservatively set as the diameter of the impacting missile (8") plus the effect of the "bending boundary layer" length. This boundary layer length is conservatively set as a multiple of twice the square root of the product of mean radius times the average thickness of two shells making up the cylindrical body of the storage overpack. From Appendix 3.B, the mean radius of the composite cylinder and the average thickness of the inner and outer shells, are~~

$$R_{\text{mean}} = 48''$$

$$T = .5 \times (.75'' + 1.25'') = 1''$$

The bending boundary layer " $\beta$ " in a shell is generally accepted to be given as  $(2(R_{\text{mean}}T)^{1/2}) = 13.85''$  for this configuration. That is, the effect of a concentrated load is resisted mainly in a length along the shell equal to the bending boundary layer. For a strike away from the ends of the shell, a boundary layer length above and below the strike location would be effective (i.e., double the boundary layer length). However, to conservatively account for resistance above and below the location of the strike, this calculated result is only increased by 1.5 in the following analysis (rather than 2). Therefore, the effective width of ring is assumed as:

$$13.85'' \times 1.5 + 8'' = 28.78''$$

The solution for ~~the 45g tipover event case 1 in Appendix 3.B~~ (performed for a unit ring width and a load of 56,184 lb.) is directly applicable if we multiply all stress and displacement results by the factor "Y" where

$$Y = (1''/28.78'') \times (843,000 \text{ lb.}/56,184 \text{ lb.}) = 0.521$$



Using this factor on the solution in Appendix 3-B, (Attachment B-1, Case 15.16) gives the following bounding results for maximum circumferential stresses (without regard for sign and location of the stress) and deformations due to the postulated tornado missile strike on the side of the storage overpack outer shell:

Maximum circumferential stress due to bending moment =  $(29,310 \text{ psi} \times Y) = 15,271 \text{ psi}$

Maximum circumferential stress due to mean tangential force =  $(18,900 \text{ lb./2 sq.inch}) \times Y = 4,923 \text{ psi}$

Change in diameter in the direction of the load =  $-0.11'' \times Y = -0.057''$

Change in diameter perpendicular to the direction of the load =  $+0.06'' \times Y = 0.031''$

Based on the above calculation, the safety factor on maximum stress for this condition is

$SF = 39,750\text{psi}/15,271 \text{ psi} = 2.60$

The allowable stress for the above calculation is the Level D membrane stress intensity limit from Table 3.1.12. This is a conservative result since the stress intensity is localized and need not be compared to primary membrane stress intensity. Even with the overestimate of impact strike force used in the calculations here ~~and in Appendix 3-G~~, the stresses remain elastic and the calculated diameter changes are small and do not prevent ready retrievability of the MPC. Note that because the stresses remain in the elastic range, there will be no post-strike permanent deformation of the inner shell.

### 3.4.8.2 HI-TRAC Transfer Cask

#### 3.4.8.2.1 Intermediate Missile Strike

HI-TRAC is always held by the handling system while in a vertical orientation completely outside of the fuel building (see Chapter 2 and Chapter 8). Therefore, considerations of instability due to a tornado missile strike are not applicable. However, the structural implications of a missile strike require consideration.

The penetration potential of the 8" missile strike on HI-TRAC (Load Case 04 in Table 3.1.5) is examined ~~at two locations in Appendix 3-H. Two locations are examined:~~

1. the lead backed outer shell of HI-TRAC.
2. the flat transfer lid consisting of multiple steel plates with a layer of lead backing.

In each case, it is shown that there is no penetration consequence that would lead to a radiological release. The following ~~paragraphs results~~ summarize the ~~analysis results~~ analyses in Appendix 3-H.

- a. The small missile will dent any surface it impacts, but no significant puncture force is generated.

- b. The following table summarizes the denting and penetration analysis performed for the intermediate missile in Appendix 3.H. Denting connotes a local deformation mode encompassing material beyond the impacting missile envelope, while penetration connotes a plug type failure mechanism involving only the target material immediately under the impacting missile. Where there is through-thickness penetration, it is shown in Appendix 3.H that the lead and the inner plate absorb any residual energy remaining after penetration of the outer plate in the 100 Ton HI-TRAC transfer lid. Both the HI-TRAC 125 and HI-TRAC 100 transfer casks are evaluated in Appendix 3.H. The table summarizes the bounding results for both transfer casks.

Location	Denting (in.)	Thru-Thickness Penetration
Outer Shell - lead backed	0.498	No (<1.0 in.)
Outer Transfer Lid Door	0.516	No (<0.75 in.) (HI-TRAC 125) Yes (>0.5 in.) (HI-TRAC 100)

The 8” missile will not penetrate the pool lid for the HI-TRAC 125D because it has a thicker bottom plate than the HI-TRAC 125 transfer lid door. In addition, the results for the 8” missile strike on the HI-TRAC outer shell are valid for the HI-TRAC 125D since all three transfer casks have the same outer shell thickness.

While the transfer cask is being transported in a horizontal orientation, the MPC lid is exposed. We conservatively assume no protective plate in place during this transport operation and evaluate the capacity of the lid peripheral groove weld to resist the impact load. The *calculated* result of calculations in Appendix 3.H, conservatively based on a reduced 5/8” weld, is as follows:

HI-TRAC MISSILE IMPACT - Capacity Results			
Item	Value (lb)	Capacity (lb)	Safety Factor = Capacity/Value
Top Lid Weld	2,262,000	2,789,000	1.23

The final calculation in this subsection is an evaluation of the circumferential stress and deformation consequences of the horizontal missile strike on the periphery of the HI-TRAC shell. It is assumed that the HI-TRAC is simply supported at its ends (while in transit) and is subject to a direct impact from the 8” diameter missile. To compute stresses, an estimate of the peak impact force is required. The effect of the water jacket to aid in the dissipation of the impact force is conservatively neglected. The only portion of the HI-TRAC cylindrical body that is assumed to resist the impact load is the two metal shells. The lead is assumed only to act as a separator to maintain the spacing

between the shells. The previous results from the lead slump analysis demonstrate that this conservative assumption on the behavior of the lead is valid. The peak value of the impact force is a function of the stiffness of the target. The target stiffness in this postulated event has the following contributions to the stiffness of the structure.

- a. a global stiffness based on a beam deformation mode, and
- b. a local stiffness based on a shell deformation mode

~~Appendix 3-Z contains information on the two transfer casks that permit the calculation of a global spring constant (i.e. the inverse of the global deflection of the cask body as a beam under a unit concentrated load). This~~ *The global spring constant (i.e., the inverse of the global deflection of the cask body as a beam under a unit concentrated load), however,* is a function of location of the strike along the length of the cask. The spring constant value varies from a minimum for a strike at the half-height to a maximum value for a strike near the supports (the trunnions). Since the peak impact force is larger for larger stiffness, it is conservative to maximize the spring constant value. Therefore, in the calculation, we neglect this spring constant for the computation of peak impact force and focus only on the spring constant arising from the local deformation as a shell, in the immediate vicinity of the strike. To this end, the spring constant is estimated by considering the three-dimensional effects of the shell solution to be replaced by the two-dimensional action of a wide ring. The width of the ring is equal to the “bending boundary layer” length on either side of the strike location plus the diameter of the striking missile. Following the analysis methodology already utilized subsection 3.4.8.1, the following information is obtained ~~from Appendix 3-AM:~~

The mean radius of the composite cylinder and the average thickness of the inner and outer shells, are (use the 100 Ton HI-TRAC data since it provides an upper bound on stress and deformation):

$$R_{\text{mean}} = 36.893$$

$$T = .5 \times (.75'' + 1.00'') = 0.875''$$

The bending boundary layer “ $\beta$ ” in a shell is generally accepted to be given as  $(2(R_{\text{mean}}T))^{1/2}$ . To account for resistance above and below the location of the strike, this calculated result is conservatively increased by multiplying by 1.5. Therefore, the effective width of ring is:

$$11.22'' \times 1.5 + 8'' = 24.84''$$

~~Appendix 3-AM contains a ring analysis of~~ *The missile impact is modeled as a point load, acting on the ring,* of magnitude equal to  $P_i = 20,570$  lb. The use of a point load in the analysis is conservative in that it overemphasizes the local stress. The actual strike area is an 8” diameter circle (or larger, if the effect of the water jacket were included).

The force is assumed resisted by inertia forces in the ring section. From the results ~~in Appendix 3-AM,~~ a spring constant can be defined as the applied load divided by the change in diameter of the ring section in the direction of the applied load. ~~Using the configuration and results in Appendix 3-AM~~ *Based on this approach,* the following local spring constant is obtained:

$$K = \pi/D1_H = \pi/0.019'' = 1,083,000 \text{ lb./inch}$$

To determine the peak impact force, a dynamic analysis of a two-body system has been performed using the “Working Model” dynamic simulation code. A two mass-spring damper system is considered with the defined spring constant representing the ring deformation effect. Figure 3.4.24 shows the results from the dynamic analysis of the impact using the computer code “Working Model”. The small square mass represents the missile, while the larger mass represents the portion of the HI-TRAC “ring” assumed to participate in the local impact. The missile weight is 275.5 lb. and the participating HI-TRAC weight is set to the weight of the equivalent ring used to determine the spring constant.

The peak impact force that results in each of the two springs used to simulate the local elasticity of the HI-TRAC (ring) is:

$$F(\text{spring}) = 124,400 \text{ lb.}$$

Since there are two springs in the model, the total impact force is:

$$P(\text{impact}) = 248,800 \text{ lb.}$$

To estimate circumferential behavior of the ring under the impact, the *previous* solution in Appendix 3-AM (using a load of 20,570 lb.) is used and amplified by the factor “Z”, where:

$$Z = 248,800 \text{ lb./}20,570 \text{ lb.} = 12.095$$

From Appendix 3-AM Consequently, the maximum circumferential stress due to the ring moment, away from the impact location, is:

$$3,037 \text{ psi} \times (69,260 \text{ in-lb/}180,900 \text{ in-lb}) \times Z = 14,230 \text{ psi}$$

At the same location, the mean stress adds an additional component (Appendix 3-AM gives the mean tangential force in the ring; the ring area is computed based on the effective width of the ring).

$$(5,143 \text{ lb./}43.47 \text{ sq.in}) \times Z = 1431 \text{ psi}$$

Therefore, the safety factor on circumferential stress causing ovalization of an effective ring section that is assumed to resist the impact is:

$$SF(\text{ring stress}) = 39,750 \text{ psi}/(1431 \text{ psi} + 14,230 \text{ psi}) = 2.54$$

The allowable stress for this safety factor calculation is obtained from Table 3.1.12 for primary membrane stress intensity for a Level D event at 350 degrees F material temperature. Noting that the actual circumferential stress in the ring remains in the elastic range, it is concluded that the MPC remains readily retrievable after the impact since there is no permanent ovalization of the cavity after the event. As noted previously, the presence of the water jacket adds an additional structural barrier that has been conservatively neglected in this analysis.

### 3.4.8.2.2 Large Missile Strike

The effects of a large tornado missile strike on the side (water jacket outer enclosure) of a loaded HI-TRAC has been simulated using a transient finite element model of the transfer cask and loaded MPC. The transient finite element code LSDYNA3D has been used (approved by the NRC for use in impact analysis (see Appendix 3.A, reference [3.A.4] for the benchmarking of this computer code)). An evaluation of MPC retrievability and global stress state (away from the impact area) are of primary interest. The finite element model includes the loaded MPC, the HI-TRAC inner and outer shells, the HI-TRAC water jacket, the lead shielding, and the appropriate HI-TRAC lids. The water in the water jacket has been neglected for conservatism in the results. The large tornado missile has been simulated by an impact force-time pulse applied on an area representing the frontal area of an 1800-kg. vehicle. The force-time data used has been previously approved by the USNRC (Bechtel Topical Report BC-TOP-9A, "Design of Structures for Missile Impact", Revision 2, 9/1974). The frontal impact area used in the finite element analysis is that area recommended in NUREG-0800, SRP 3.5.1.4, Revision 2, 1981).

~~Appendix 3-AN describes the finite element model, the input data used, and provides graphical results necessary to the evaluation of retrievability and state of stress. A summary of the results from Appendix 3-AN is presented below for the HI-TRAC 100 and HI-TRAC 125 transfer casks. Since the dimensions of the inner shell, the outer shell, the lead shielding, and the water jacket enclosure panels are the same in both the HI-TRAC 125 and the HI-TRAC 125D, the results from the HI-TRAC 125 are considered accurate for the HI-TRAC 125D. The allowable value listed for the stress intensity for this Level D event comes from Table 3.1.17.~~

The results from the dynamic analysis have been summarized below.

<b>SUMMARY OF RESULTS FROM LARGE TORNADO MISSILE IMPACT ANALYSIS</b>		
<b>ITEM – HI-TRAC 100</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Maximum Stress Intensity in Water Jacket (ksi)	28.331	58.7
Maximum Stress Intensity in Inner Shell (ksi)	11.467	58.7
Maximum Plastic Strain in Water Jacket	0.0000932	-
Maximum Plastic Strain in Inner Shell	0.0	-

<b>ITEM – HI-TRAC 125</b>	<b>CALCULATED VALUE</b>	<b>ALLOWABLE VALUE</b>
Maximum Stress Intensity in Water Jacket (ksi)	19.073	58.7
Maximum Stress Intensity in Inner Shell (ksi)	6.023	58.7
Maximum Plastic Strain in Water Jacket	0.0	-
Maximum Plastic Strain in Inner Shell	0.0	-

The above results demonstrate that:

1. The retrievability of the MPC in the wake of a large tornado missile strike is not adversely affected since the inner shell does not experience any plastic deformation.
2. The maximum primary stress intensity, away from the impact interface on the HI-TRAC water jacket, is below the applicable ASME Code Level D allowable limit for NF, Class 3 structures.

### 3.4.9 HI-TRAC Drop Events (Load Case 02.b in Table 3.1.5)

During transit, the HI-TRAC 125 or HI-TRAC 100 transfer cask may be carried horizontally with the transfer lid in place. Analyses have been performed to demonstrate that under a postulated carry height; the design basis 45g deceleration is not exceeded. The analyses have been performed using two different simulation models. A simplified model of the drop event is performed using the computer simulation code “Working Model 2D”. The analysis using “Working Model 2D” assumed the HI-TRAC and the contained MPC acted as a single rigid body. A second model of the drop event uses DYNA3D, considers the multi-body analysis of HI-TRAC and the contained MPC as individual bodies, and is finite element based. In what follows, we outline the problem and the results obtained using each solution methodology.

#### 3.4.9.1 Working Model 2D Analysis of Drop Event

The analysis model conservatively neglects all energy absorption by any component of HI-TRAC; all kinetic energy is transferred to the ground through the spring-dampers that simulate the foundation (ground). If the HI-TRAC suffers a handling accident causing a side drop to the ground, impact will only occur at the top and bottom ends of the vessel. The so-called “hard points” are the top end lifting trunnions, the bottom end rotation trunnions, and the projecting ends of the transfer lid. Noting that the projecting hard points are of different dimensions and will impact the target at different times because of the HI-TRAC geometry, any simulation model must allow for this possibility.

A dynamic analysis of a horizontal drop, with the lowest point on the HI-TRAC assumed 50” above the surface of the target (larger than the design basis limit of 42”), is considered in Appendix 3-Z for the HI-TRAC 125 and for the HI-TRAC 100. Figure 3.4.22 shows the transfer cask orientation. The HI-TRAC is considered as a rigid body (Appendix 3-Z contains calculations that demonstrate that the lowest beam mode frequency is well above 33 Hz so that no dynamic amplification need be included). The effects of the ISFSI pad and the underlying soil are included using a simple spring-damper model based on a static classical Theory of Elasticity solution. The “worst” orientation of a horizontally carried HI-TRAC with the transfer cask impacting an elastic surface is considered. The HI-TRAC is assumed to initially impact the target with the impact force occurring over the rectangular surface of the transfer lid (11.875” x 81”). “Worst” is defined here as meaning an impact at a location having the maximum value of an elastic spring constant simulating the resistance of the target interface. Appendix 3-AL provides the calculation of the elastic spring-damper that simulates the contact spring. The geometry and material properties used in Appendix 3-AL reflect the USNRC accepted reference pad and soil (Table 2.2.9 - the pad thickness used is 36” and the Young’s Modulus of the elastic soil is the upper limit value E=28,000 psi). The use of an elastic representation of the target surface is conservative as it minimizes the energy absorption capacity of the target and maximizes the deceleration loads developed during the impact. Also considered in Appendix 3-AL is a calculation of  $t$ . The spring constant is also calculated based on an assumption that impact at the lower end of HI-TRAC first occurs at the pocket trunnion. The results in Appendix 3-AL demonstrate that this spring constant is lower and therefore would lead to a lower impact force. Therefore, the dynamic analysis of the handling accident is performed assuming initial impact with the flat rectangular short end of the transfer lid. Subsequent to the initial impact, the HI-TRAC rotates in accordance with the dynamic equations of equilibrium and a secondary impact at the top of the transfer cask occurs. The impact is at the edge of the water jacket.

The following table summarizes the results from the dynamic analyses (using the Working Model 2D computer code) documented in Appendix 3-Z:

<b>HI-TRAC Handling Analysis – Working Model Analysis of Horizontal Drop</b>			
<b>Item</b>	<b>Value</b>	<b>Allowable</b>	<b>Safety Factor</b>
HI-TRAC 125 – Primary Impact Deceleration (g’s)	32.66	45	1.38
HI-TRAC 125 – Secondary Impact Deceleration (g’s)	26.73	45	1.68
HI-TRAC 100 – Primary Impact Deceleration (g’s)	33.18	45	1.36
HI-TRAC 100 – Secondary Impact Deceleration (g’s)	27.04	45	1.66
Axial Membrane Stress Due to HI-TRAC 125 Bending as a Beam - Level D Drop (psi)	19.06	39.75	2.085
Axial Membrane Stress Due to HI-TRAC 100 Bending as a Beam - Level D Drop (psi)	15.77	39.75	2.52

In the table above, the decelerations are measured at points corresponding to the base and top of the fuel assemblies contained inside the MPC. The dynamic drop analysis reported above, using the Working Model 2D rigid body-spring model proved that decelerations are below the design basis value and that global stresses were within allowable limits.

#### 3.4.9.2 DYNA3D Analysis of Drop Event

An independent evaluation of the drop event to delineate the effect of target non-linearity and the flexibility of the transfer cask has been performed using DYNA3D. ~~Appendix 3.AN provides details of the HI-TRAC drop model, the data input, and extensive graphical results.~~ Both the HI-TRAC 125 and HI-TRAC 100 transfer casks are modeled as part of the cask-pad-soil interaction finite element model set forth in NUREG/CR-6608 and validated by an NRC reviewed and approved Holtec topical report (see reference [3.A.4] in Appendix 3.A). The model uses the identical MPC and target pad/soil models employed in the accident analyses of the HI-STORM 100 overpack. The HI-TRAC inner and outer shells, the contained lead, the transfer lid, the water jacket metal structure, and the top lids are included in the model. The water jacket is assumed empty for conservatism.

Two side drop orientations are considered (see Figures 3.4.27 and 3.4.28). The first drop assumes that the plane of the lifting and rotation trunnions is horizontal with primary impact on the short side of the transfer lid. This maximizes the angle of slapdown, and represents a credible drop configuration where the HI-TRAC cask is dropped while being carried horizontally. The second drop orientation assumes primary impact on the rotation trunnion and maximizes the potential for the lifting trunnion to participate in the secondary impact. This is a non-credible event that assumes complete separation from the transfer vehicle and a ninety-degree rotation prior to impact. Nevertheless, it is the only configuration where the trunnions could be involved in both primary and secondary impacts.

For each simulation performed, the lowest point on the HI-TRAC cask (either the transfer lid edge or the rotation trunnion) is set at 42" above the target interface. Decelerations are measured at the top lid, the cask centroidal position, and the transfer lid. Normal forces were measured at the primary impact interface, at the secondary impact interface, and at the top lid/MPC interface. Decelerations are filtered at 350 Hz.



The following key results summarize the analyses documented in the new Appendix 3.AN:

ITEM	HI-TRAC 125		HI-TRAC 100		ALLOWABLE
	Horizontal	Vertical	Horizontal	Vertical	
Max. Top Lid Vertical Deceleration – Secondary Impact (g's)	25.5	32	36.5	45 <sup>†</sup>	45
Centroid Vertical Deceleration – at Time of Secondary Impact (g's)	9.0	13.0	10.0	17.5	45
Max. Transfer Lid Vertical Deceleration – Primary Impact (g's)	30.8	23.5	35.0	31.75	45
Maximum Normal Force at Primary Impact Site (kips)	1,950.	1,700	1,700	1,700	-
Maximum Normal Force at Secondary Impact Site (kips)	1,300.	1,850.	1,500.	1,450.	-
Maximum MPC/Top Lid Interface Force (kips)	132.	-	39.	-	-
Maximum Diametral Change of Inner Shell (inch)	0.228	0.113	Not Computed	0.067	0.3725
Maximum Von Mises Stress (ksi)	37.577	38.367	40.690	40.444	58.7*

<sup>†</sup> The deceleration at the top of the basket is estimated at 41 g's

\* Allowable Level D Stress Intensity for Primary Plus Secondary Stress Intensity

The results presented in Appendix 3.AN and summarized above demonstrate that both the HI-TRAC 125 and HI-TRAC 100 transfer casks are sufficiently robust to perform their function during and after the postulated handling accidents. We also note that the results, using the Working Model single rigid body dynamic model (see Subsection 3.4.9.1), are in reasonable agreement with the results predicted by the DYNA3D multi-body finite element dynamic model although performed for a different drop height with deceleration measurements at different locations on the HI-TRAC.

The results reported above for maximum interface force at the top lid/MPC interface are used as input to a separate analysis, which in Appendix 3.AH demonstrates that the top lid contains the MPC during and after a handling accident. The results reported above for the maximum normal force at the primary impact site (the transfer lid) have been used to calculate the maximum interface force at the bottom flange/transfer lid interface. This result is needed to insure that the interface input forces used in Appendices 3.AD and 3.AJ to evaluate transfer lid separation are indeed bounding. To obtain the interface force between the HI-TRAC transfer lid and the HI-TRAC bottom flange, it is sufficient to take a free-body of the transfer lid and write

the dynamic force equilibrium equation for the lid. Figure 3.4.29 shows the free body with appropriate notation. The equation of equilibrium is:

$$M_{TL} a_{TL} = F_I - G_I$$

where

$M_{TL}$  = the mass of the transfer lid

$a_{TL}$  = the time varying acceleration of the centroid of the transfer lid

$F_I$  = the time varying contact force at the interface with the target

$G_I$  = the time varying interface force at the bottom flange/transfer lid interface

Solving for the interface force give the result

$$G_I = F_I - M_{TL} a_{TL}$$

Using the appropriate transfer lid mass and acceleration, together with the target interface force at the limiting time instant, provides values for the interface force. The table below provides the results of this calculation for the HI-TRAC 125 and HI-TRAC 100 transfer casks.

<b>Item</b>	<b>Calculated from Equilibrium (kips)</b>
HI-TRAC 125 – Trunnions Horizontal	1,183.
HI-TRAC 125 – Trunnions Vertical	1,272.
HI-TRAC 100 – Trunnions Horizontal	1,129.
HI-TRAC 100 – Trunnions Vertical	1,070.

### 3.4.9.3 Horizontal Drop of HI-TRAC 125D

The previous subsection addressed the 42” horizontal drop of the HI-TRAC 125 and HI-TRAC 100, including an evaluation of the bolted connection between the transfer lid, which sustains the primary impact, and the cylindrical body of the loaded HI-TRAC. The HI-TRAC 125D does not have a bolted connection between the bottom flange and the cylindrical body of the cask. However, the transverse protrusions (bottom flange, lifting trunnions, and optional attachment lugs/support tabs at the top of the cask) spawn different impact scenarios. The uncontrolled lowering of the cask is assumed to occur from a height of 42” measured to the lowest location on the HI-TRAC 125D in the horizontal orientation.

The maximum decelerations for the HI-TRAC 125D are comparable to the drop results for the HI-TRAC 125 when the plane of the lifting and rotation trunnions is vertical. Although the HI-TRAC 125D has no rotation trunnions, its bottom flange extends radially beyond the water jacket shell by approximately the same amount as the HI-TRAC 125 rotation trunnions and thereby establishes a similar “hard point” for primary impact in terms of distance from the cask centerline. More important, because the bottom flange is positioned closer to the base of the HI-TRAC 125D than the rotation trunnions are in the HI-TRAC 125, the slap-down angle for the HI-TRAC 125D is less. The shallower angle decreases the participation of the lifting trunnion during the secondary impact, and increases the participation of the water jacket shell. Since the water jacket shell is a more flexible structure than the lifting trunnion, the deceleration of the HI-TRAC 125D cask during secondary impact is slightly less than the calculated deceleration of the HI-TRAC 125. In the HI-TRAC 125D, there is no bolted connection at the bottom flange/cask body interface that is active in load transfer from the flange to the cask body. It is therefore concluded that this drop scenario for the HI-TRAC 125D is bounded by the similar evaluation for the HI-TRAC 125.

A second HI-TRAC 125D drop scenario with two attachment lugs/support tabs in a vertical plane is the most limiting scenario. The tab dimensions are such that primary impact occurs at the top end of the cask when the support tabs impact the target surface, followed by a slap-down and a secondary impact at the bottom flange.

The evaluation of HI-TRAC 125D drop scenario is performed using the computer code Working Model 3D (WM) (now known as Visual Nastran Desktop). First, the WM code is used to simulate the “Scenario A” drop of the HI-TRAC 125 in order to establish appropriate parameters to “benchmark” WM against the DYNA3D solution. The table below summarizes the results of the Working Model/DYNA3D benchmark comparison (the DYNA3D solution for the HI-TRAC 125 (Scenario A) is documented in Appendix 3.AN). Figure 3.4.48 shows the benchmark configuration after the drop event.

Comparison of HI-TRAC 125 Drop Results (Scenario A)		
	DYNA3D	Working Model
Vertical Deceleration of Top Lid (secondary impact) g's	32	33.49
Vertical Deceleration at Bottom Lid (primary impact on rotation trunnion) g's	23.5	23.59

The benchmarked Working Model simulation was then modified to simulate the second drop scenario of the HI-TRAC 125D with support tabs in a vertical plane; primary impact now occurred at the top end with secondary impact at the bottom flange. Figure 3.4.49 shows the configuration of the HI-TRAC 125D after this scenario. The impact parameters were unchanged from the benchmark model except for location. The acceleration results from the 42” horizontal drop of the HI-TRAC 125D in this second *drop scenario* are summarized below.

Results From HI-TRAC 125D 42" Drop	
Vertical Deceleration of Top Lid (primary impact on support tab) g's	36.75
Vertical Deceleration of Pool Lid (secondary impact on bottom flange) g's	29.27

The resulting g loads at the top of the active fuel region for the HI-TRAC 125D, with primary impact on the support tabs, are increased over the loads computed for the HI-TRAC 125 but remain well below the design basis limit.

3.4.10 HI-STORM 100 Non-Mechanistic Tip-over and Vertical Drop Event (Load Cases 02.a and 02.c in Table 3.1.5)

Pursuant to the provision in NUREG-1536, a non-mechanistic tip-over of a loaded HI-STORM 100 System on to the ISFSI pad is considered in this report. Analyses are also performed to determine the maximum deceleration sustained by a vertical free fall of a loaded HI-STORM 100 System from an 11" height onto the ISFSI pad. The objective of the analyses is to demonstrate that the plastic deformation in the fuel basket is sufficiently limited to permit the stored SNF to be retrieved by normal means, does not have an adverse effect on criticality safety, and that there is no significant loss of radiation shielding in the system.

Ready retrievability of the fuel is presumed to be ensured: if global stress levels in the MPC structure meet Level D stress limits during the postulated drop events; if any plastic deformations are localized; and if no significant permanent ovalization of the overpack into the MPC envelope space, remains after the event.

Subsequent to the accident events, the storage overpack must be shown to contain the shielding so that unacceptable radiation levels do not result from the accident.

Appendix 3.A provides a description of the dynamic finite element analyses undertaken to establish the decelerations resulting from the postulated event. A non-mechanistic tip-over is considered together with an end drop of a loaded HI-STORM 100 System. A dynamic finite element analysis of each event is performed using a commercial finite element code well suited for such dynamic analyses with interface impact and non-linear material behavior. This code and methodology have been fully benchmarked against Lawrence Livermore Laboratories test data and correlation [3.4.12].

The table below provides the values of computed peak decelerations at the top of the fuel basket for the vertical drop and the non-mechanistic tipover scenarios. It is seen that the peak deceleration is below 45 g's.

### Filtered Results for Drop and Tip-Over Scenarios for HI-STORM

Drop Event	Max. Deceleration at the Top of the Basket (g's)	
	Set A(36'' Thick Pad)	Set B(28'' Thick Pad)
End Drop for 11 inches	43.98	41.53
Non-Mechanistic Tip-over	42.85	39.91

The tipover analysis performed in Appendix 3.A is based on the HI-STORM 100 geometry and a bounding weight. The fact that the HI-STORM 100S(232) is shorter and has a lower center of gravity suggests that the impact kinetic energy is reduced so that the target would absorb the energy with a lower maximum deceleration. However, since the actual weight of a HI-STORM 100S(232) is less than that of a HI-STORM 100 by a significant amount, the predicted maximum rigid body deceleration would tend to increase slightly. Since there are two competing mechanisms at work, it is not a foregone conclusion that the maximum rigid body deceleration level is, in fact, reduced if a HI-STORM 100S(232) suffers a non-mechanistic tipover onto the identical target as the HI-STORM 100. The situation is clearer for the HI-STORM 100S(243), which is virtually equal in weight to the HI-STORM 100, yet its center of gravity when loaded is almost one inch lower. In what follows, we present a summary of the analysis undertaken to demonstrate conclusively that the result for maximum deceleration level in the HI-STORM 100 tipover event does bound the corresponding value for the HI-STORM 100S(232), and, therefore, we need only perform a detailed dynamic finite element analysis for the HI-STORM 100.

Appendix 3.A presents a result for the angular velocity of the cylindrical body representing a HI-STORM 100 just prior to impact with the defined target. The result is expressed in Subsection 3.A.6 in terms of the cask geometry, and the ratio of the mass divided by the mass moment of inertia about the corner point that serves as the rotation origin. Since the mass moment of inertia is also linearly related to the mass, the angular velocity at the instant just prior to target contact is independent of the cask mass. Subsequent to target impact, we investigate post-impact response by considering the cask as a cylinder rotating into a target that provides a resistance force that varies linearly with distance from the rotation point. We measure “time” as starting at the instant of impact, and develop a one-degree-of-freedom equation for the post-impact response (for the rotation angle into the target) as:

$$\ddot{\theta} + \omega^2\theta = 0$$

where

$$\omega^2 = \frac{kL^3}{3I_A}$$

The initial conditions at time=0 are: the initial angle is zero and the initial angular velocity is equal to the rigid body angular velocity acquired by the tipover from the center-of-gravity over corner position. In the above relation, L is the length of the overpack, I is the mass moment of inertia defined in Appendix 3.A, and k is a “spring constant” associated with the target resistance. If we solve for the maximum angular acceleration subsequent to time =0, we obtain the result in terms of the initial angular velocity as:

$$\ddot{\theta}_{\max} = \omega \dot{\theta}_0$$

If we form the maximum linear acceleration at the top of the four-inch thick lid of the overpack, we can finally relate the decelerations of the HI-STORM 100 and the HI-STORM 100S(232) solely in terms of their geometry properties and their mass ratio. The value of “k”, the target spring rate is the same for both overpacks so it does not appear in the relationship between the two decelerations. After substituting the appropriate geometry and calculated masses, we determine that the ratio of maximum rigid body decelerations at the top surface of the four-inch thick top lid plates is:

$$A_{\text{HI-STORM 100S(232)}}/A_{\text{HI-STORM 100}} = 0.946$$

Therefore, as postulated, there is no need to perform a separate DYNA3D analysis for the HI-STORM 100S hypothetical tipover.

~~Appendix 3.B contains a~~ A simple elastic strength of materials calculation *is performed* to demonstrate that the cylindrical storage overpack will not permanently deform to the extent that the MPC cannot be removed by normal means after a tip-over event. ~~It is demonstrated in that appendix~~ *The results demonstrate* that the maximum diametrical closure of the cylindrical cavity is less than the initial clearance between the overpack MPC support channels and the MPC canister. Primary circumferential membrane stresses in the MPC shell remain in the elastic range during a tip-over (see Table 3.4.6 summary safety factors); therefore, no permanent global ovalization of the MPC shell occurs as a result of the drop.

To demonstrate that the shielding material will continue to perform its function after a tip-over accident, the stress and strain levels in the metal components of the storage overpack are examined at the end of the tip-over event. The results obtained in Appendix 3.A for impact decelerations conservatively assumed a rigid storage overpack model to concentrate nearly all energy loss in the target. However, to assess the state of stress and strain in the storage overpack after an accident causing a tip-over, the tip-over analysis was also performed using a non-rigid storage overpack model using overpack material properties listed in Appendix 3.A. Figure 3.4.13 shows the calculated von Mises stress in the top lid and outer shell at 0.08 seconds after the initiation of impact. Figure 3.4.14 shows the residual plastic strains in the same components. Figures 3.4.15 and 3.4.16 provide similar results for the inner shell, the radial plates, and the support channels<sup>†</sup>. The results show that while some plastic straining occurs, accompanied by stress levels above the yield stress of the

<sup>†</sup> During fabrication the channels are attached to the inner shell by one of two methods, either the channels are welded directly to the inner shell or they are welded to a pair of L-shaped angles (i.e., channel mounts) that are pre-fastened to the inner shell. The results presented in Figures 3.4.16a and 3.4.16b bound the results from both methods of attachment.

material, there is no tearing in the metal structure which confines the radiation shielding (concrete). Therefore, there is no gross failure of the metal shells enclosing the concrete. The shielding concrete will remain inside the confines of the storage overpack and maintain its performance after the tipover event.

#### 3.4.11 Storage Overpack and HI-TRAC Transfer Cask Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years; therefore, the License Life (please see glossary) of all components is 20 years. Nonetheless, the HI-STORM 100 and 100S Storage overpacks and the HI-TRAC transfer cask are engineered for 40 years of design life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. In addition, the storage overpack and HI-TRAC are designed, fabricated, and inspected under the comprehensive Quality Assurance Program discussed in Chapter 13 and in accordance with the applicable requirements of the ACI and ASME Codes. This assures high design margins, high quality fabrication, and verification of compliance through rigorous inspection and testing, as describe in Chapter 9 and the design drawings in Section 1.5. Technical Specifications defined in Chapter 12 assure that the integrity of the cask and the contained MPC are maintained throughout the components' design life. The design life of a component, as defined in the Glossary, is the minimum duration for which the equipment or system is engineered to perform its intended function if operated and maintained in accordance with the FSAR. The design life is essentially the lower bound value of the service life, which is the expected functioning life of the component or system. Therefore, component longevity should be: licensed life < design life < service life. (The licensed life, enunciated by the USNRC, is the most pessimistic estimate of a component's life span.) For purposes of further discussion, we principally focus on the service life of the HI-STORM 100 System components that, as stated earlier, is the reasonable expectation of equipment's functioning life span.

The service life of the storage overpack and HI-TRAC transfer cask is further discussed in the following sections.

##### 3.4.11.1 Storage Overpack

The principal design considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

##### Exposure to Environmental Effects

In the following text, all references to HI-STORM 100 also apply to HI-STORM 100S. All exposed surfaces of HI-STORM 100 are made from ferritic steels that are readily painted. Concrete, which serves strictly as a shielding material, is completely encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete, are ruled out for HI-STORM 100. Under normal storage conditions, the bulk temperature of the HI-STORM 100 storage overpack will, because of its large thermal inertia, change very gradually with time. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM 100 storage overpack. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM 100 because HI-STORM 100 does not

rely on rebars (indeed, it contains no rebars). As discussed in Appendix 1.D, the aggregates, cement and water used in the storage cask concrete are carefully controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation. In addition, the storage overpack is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the 40-year design life of the storage overpack as defined in Subsection 2.2.3 and evaluated in Chapter 11.

### Material Degradation

The relatively low neutron flux to which the storage overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The controlled environment of the ISFSI storage pad mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage overpack throughout the 40-year design life are defined in Chapter 9. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STORM 100 System is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [3.4.11], which concluded that dry storage systems designed, fabricated, inspected, and operate in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

#### 3.4.11.2 Transfer Cask

The principal design considerations that bear on the adequacy of the HI-TRAC Transfer Cask for the service life are addressed as follows:

#### Exposure to Environmental Effects

All transfer cask materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The HI-TRAC is designed for repeated normal condition handling operations with high factor of safety, particularly for the lifting trunnions, to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the trunnion material, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture



during the lowest temperature permitted for loading, as discussed in Chapter 12.

### Material Degradation

All transfer cask materials that are susceptible to corrosion are coated. The controlled environment in which the HI-TRAC is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the HI-TRAC materials are subjected do not result in radiation embrittlement or degradation of the HI-TRAC's shielding materials that could impair the HI-TRAC's intended safety function. The HI-TRAC transfer cask materials are selected for durability and wear resistance for their deployment.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the HI-TRAC transfer cask throughout the 40-year design life are defined in Chapter 9. These requirements include provisions for routine inspection of the HI-TRAC transfer cask for damage prior to each use, including an annual inspection of the lifting trunnions. Precautions are taken during lid handling operations to protect the sealing surfaces of the pool lid. The leak tightness of the liquid neutron shield is verified periodically. The water jacket pressure relief valves and other fittings used can be easily removed.

#### 3.4.12 MPC Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC (i.e., licensed life) is 20 years. Nonetheless, the HI-STORM 100 MPC is designed for 40 years of design life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. Additional assurance of the integrity of the MPC and the contained SNF assemblies throughout the 40-year life of the MPC is provided through the following:

- Design, fabrication, and inspection in accordance with the applicable requirements of the ASME Code as described in Chapter 2 assures high design margins.
- Fabrication and inspection performed in accordance with the comprehensive Quality Assurance program discussed in Chapter 13 assures competent compliance with the fabrication requirements.
- Use of materials with known characteristics, verified through rigorous inspection and testing, as described in Chapter 9, assures component compliance with design requirements.
- Use of welding procedures in full compliance with Section III of the ASME Code ensures high-quality weld joints.

Technical Specifications, as defined in Chapter 12, have been developed and imposed on the MPC that assure that the integrity of the MPC and the contained SNF assemblies are maintained throughout the 40-year design life of the MPC.

The principal design considerations bearing on the adequacy of the MPC for the service life are summarized below.

### Corrosion

All MPC materials are fabricated from corrosion-resistant austenitic stainless steel and passivated aluminum. The corrosion-resistant characteristics of such materials for dry SNF storage canister applications, as well as the protection offered by these materials against other material degradation effects, are well established in the nuclear industry. The moisture in the MPC is removed to eliminate all oxidizing liquids and gases and the MPC cavity is backfilled with dry inert helium at the time of closure to maintain an atmosphere in the MPC that provides corrosion protection for the SNF cladding throughout the dry storage period. The preservation of this non-corrosive atmosphere is assured by the inherent seal worthiness of the MPC confinement boundary integrity (there are no gasketed joints in the MPC).

### Structural Fatigue

The passive non-cyclic nature of dry storage conditions does not subject the MPC to conditions that might lead to structural fatigue failure. Ambient temperature and insolation cycling during normal dry storage conditions and the resulting fluctuations in MPC thermal gradients and internal pressure is the only mechanism for fatigue. These low stress, high-cycle conditions cannot lead to a fatigue failure of the MPC that is made from stainless alloy stock (endurance limit well in excess of 20,000 psi). All other off-normal or postulated accident conditions are infrequent or one-time occurrences, which cannot produce fatigue failures. Finally, the MPC uses materials that are not susceptible to brittle fracture.

### Maintenance of Helium Atmosphere

The inert helium atmosphere in the MPC provides a non-oxidizing environment for the SNF cladding to assure its integrity during long-term storage. The preservation of the helium atmosphere in the MPC is assured by the robust design of the MPC confinement boundary described in Section 7.1. Maintaining an inert environment in the MPC mitigates conditions that might otherwise lead to SNF cladding failures. The required mass quantity of helium backfilled into the canister at the time of closure, as defined in the Technical Specification contained in Subsection 12.3.3, and the associated leak tightness requirements for the canister defined in the Technical Specification contained in Chapter 12, are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout the 40-year design life.

### Allowable Fuel Cladding Temperatures

The helium atmosphere in the MPC promotes heat removal and thus reduces SNF cladding temperatures during dry storage. In addition, the SNF decay heat will substantially attenuate over a 40-year dry storage period. Maintaining the fuel cladding temperatures below allowable levels during long-term dry storage mitigates the damage mechanism that might otherwise lead to SNF cladding failures. The allowable long-term SNF cladding temperatures used for thermal acceptance

of the MPC design are conservatively determined, as discussed in Section 4.3.

#### Neutron Absorber Boron Depletion

The effectiveness of the fixed borated neutron absorbing material used in the MPC fuel basket design requires that sufficient concentrations of boron be present to assure criticality safety during worst case design basis conditions over the 40-year design life of the MPC. Information on the characteristics of the borated neutron absorbing material used in the MPC fuel basket is provided in Subsection 1.2.1.3.1. The relatively low neutron flux, which will continue to decay over time, to which this borated material is subjected, does not result in significant depletion of the material's available boron to perform its intended safety function. In addition, the boron content of the material used in the criticality safety analysis is conservatively based on the minimum specified boron areal density (rather than the nominal), which is further reduced by 25% for analysis purposes, as described in Section 6.1. Analysis discussed in Section 6.3.2 demonstrates that the boron depletion in the *neutron absorber material* ~~Boral~~ is negligible over a 50-year duration. Thus, sufficient levels of boron are present in the fuel basket neutron absorbing material to maintain criticality safety functions over the 40-year design life of the MPC.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review, which concluded that dry storage systems designed, fabricated, inspected, and operated in the manner of the requirements set down in this document are adequate for a 100-year service life, while satisfying the requirements of 10CFR72.

#### 3.4.13 Design and Service Life

The discussion in the preceding sections seeks to provide the logical underpinnings for setting the design life of the storage overpacks, the HI-TRAC transfer cask, and the MPCs as forty years. Design life, as stated earlier, is a lower bound value for the expected performance life of a component (service life). If operated and maintained in accordance with this Final Safety Analysis Report, Holtec International expects the service life of its HI-STORM 100 and HI-STORM 100S components to substantially exceed their design life values.

Table 3.4.1

## FINITE ELEMENTS IN THE MPC STRUCTURAL MODELS

<b>MPC Type</b>	<b>Model Type</b>		
<b>Element Type</b>	<b>Basic</b>	<b>0 Degree Drop</b>	<b>45 Degree Drop</b>
<b>MPC-24</b>	1068	1114	1113
BEAM3	1028	1028	1028
PLANE82	0	0	0
CONTAC12	40	38	38
CONTAC26	0	45	45
COMBIN14	0	3	2
<b>MPC-32</b>	1374	1604	1603
BEAM3	1346	1346	1346
CONTAC12	28	27	24
CONTAC26	0	229	228
COMBIN14	0	2	5
<b>MPC-68</b>	1842	2066	2063
BEAM3	1782	1782	1782
PLANE82	16	16	16
CONTAC12	44	43	40
CONTAC26	0	223	222
COMBIN14	0	2	3
<b>MPC-24E</b>	1070	1124	1122
BEAM3	1030	1030	1030
PLANE82	0	0	0
CONTAC12	40	38	38
CONTAC26	0	53	52
COMBIN14	0	3	2

**TABLE 3.4.2**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

Material/Component	Fuel Pool (Borated and Unborated Water) <sup>†</sup>	ISFSI Pad (Open to Environment)
<u>Alloy X:</u>  - MPC Fuel Basket - MPC Baseplate - MPC Shell - MPC Lid - MPC Fuel Spacers	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions or interactions with spent fuel.	The MPC internal environment will be an inert (helium) atmosphere and the external surface will be exposed to ambient air. No adverse interactions identified.
<u>Aluminum:</u>  - Heat Conduction Elements	Aluminum and stainless steel form a galvanic couple. However, aluminum will be used in a passivated state. Upon passivation, aluminum forms a thin ceramic (Al <sub>2</sub> O <sub>3</sub> ) barrier. Therefore, during the short time they are exposed to pool water, significant corrosion of aluminum or production of hydrogen is not expected (see operational requirements under “ <i>Neutron Absorber Material Boral</i> ” below).	In a non-aqueous atmosphere, galvanic corrosion is not expected.
<del><u>Neutron Absorber Material Boral:</u></del>  ——— Neutron Absorber	<del>The Boral will be passivated before installation in the fuel basket to minimize the amount of hydrogen released from the aluminum-water reaction to a non-combustible concentration during MPC lid welding or cutting operations. Extensive in-pool experience on spent fuel racks with no adverse reactions. See Chapter 8 for additional requirements for combustible gas monitoring and recommended-required actions for control of combustible gas accumulation under the MPC lid.</del>	No adverse potential reactions identified.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

Material/Component	Fuel Pool (Borated and Unborated Water) <sup>†</sup>	ISFSI Pad (Open to Environment)
<u>Steels:</u> - SA350-LF2 - SA350-LF3 - SA203-E - SA515 Grade 70 - SA516 Grade 70 - SA193 Grade B7 - SA106 (HI-TRAC)	All exposed steel surfaces (except seal areas, and pocket trunnions) will be coated with paint specifically selected for performance in the operating environments. Even without coating, no adverse reactions (other than nominal corrosion) have been identified. Lid bolts are plated and the threaded portion of the bolt anchor blocks is coated to seal the threaded area.	Internal surfaces of the HI-TRAC will be painted and maintained. Exposed external surfaces (except those listed in fuel pool column) will be painted and will be maintained with a fully painted surface. No adverse reactions identified.
<u>Steels:</u> - SA516 Grade 70 - SA203-E - SA350-LF3 Storage Overpack	HI-STORM 100 storage overpack is not exposed to fuel pool environment.	Internal and external surfaces will be painted (except for bolt locations that will have protective coating). External surfaces will be maintained with a fully painted surface. No adverse reaction identified.
<u>Stainless Steels:</u> - SA240 304 - SA193 Grade B8 - 18-8 S/S  Miscellaneous Components	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions.	Stainless steel has a long proven history of corrosion resistance when exposed to the atmosphere. These materials are used for bolts and threaded inserts. No adverse reactions with steel have been identified. No impact on performance.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

<b>Material/Component</b>	<b>Fuel Pool (Borated and Unborated Water)<sup>†</sup></b>	<b>ISFSI Pad (Open to Environment)</b>
<u>Nickel Alloy:</u> - SB637-NO7718 Lifting Trunnion	No adverse reactions with borated or unborated water.	Exposed to weathering effects. No adverse reactions with storage overpack closure plate. No impact on performance.
<u>Brass/Bronze:</u> - Pressure Relief Valve HI-TRAC	Small surface of pressure relief valve will be exposed. No significant adverse impact identified.	Exposed to external weathering. No loss of function expected.
<u>Holtite-A:</u> - Solid Neutron Shield	The neutron shield is fully enclosed. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.	The neutron shield is fully enclosed in the outer enclosure. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.

**TABLE 3.4.2 (CONTINUED)**  
**HI-STORM 100 SYSTEM MATERIAL COMPATIBILITY**  
**WITH OPERATING ENVIRONMENTS**

Material/Component	Fuel Pool (Borated and Unborated Water) <sup>†</sup>	ISFSI Pad (Open to Environment)
<u>Paint:</u> <ul style="list-style-type: none"> <li>- Carboline 890</li> <li>- Thermaline 450</li> </ul>	<p>Carboline 890 used for all HI-STORM 100 surfaces and only HI-TRAC exterior surfaces. Acceptable performance for short-term exposure in mild borated pool water.</p> <p>Thermaline 450 selected for HI-TRAC internal surfaces for excellent high temperature resistance properties. Will only be exposed to demineralized water during in-pool operations as annulus is filled prior to placement in the spent fuel pool and the inflatable seal prevents fuel pool water in-leakage. No adverse interaction identified which could affect MPC/fuel assembly performance.</p>	<p>Good performance on surfaces. Discoloration is not a concern.</p>
<u>Elastomer Seals:</u>	No adverse reactions identified.	Only used during fuel pool operations.
<u>Lead:</u>	Enclosed by carbon steel. Lead is not exposed to fuel pool water. Lead has no interaction with carbon steel.	Enclosed by carbon steel. Lead is not exposed to ambient environment. Lead has no interaction with carbon steel.
<u>Concrete:</u>	Storage overpack is not exposed to fuel pool water.	Concrete is enclosed by carbon steel and not exposed to ambient environment. Concrete has no interaction with carbon steel.

<sup>†</sup> HI-TRAC/MPC short-term operating environment during loading and unloading.



**TABLE 3.4.3  
FUEL BASKET RESULTS - MINIMUM SAFETY FACTORS**

<b>Load Case I.D.</b>	<b>Loading<sup>†</sup></b>	<b>Safety Factor</b>	<b>Location in FSAR Where the Analysis is Performed</b>
F1	T, T'	No interference	<i>Subsection 3.4.4.23.I, 3.U, 3.W, 3.AF</i>
F2	D + H	2.79	3.AA of Docket 72-1008
F3			
F3.a	D + H' (end drop)	3.59	3.4.4.3.1.3
F3.b	D + H' (side drop 0 deg.)	1.32	Table 3.4.6
F3.c	D + H' (side drop 45 deg.)	1.28	Table 3.4.6

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**TABLE 3.4.4  
MPC RESULTS - MINIMUM SAFETY FACTOR**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†,††</sup></b>	<b>Safety Factor</b>	<b>Location in FSAR</b>	<b>Where the Analysis is Performed</b>
E1				
E1.a	Design internal pressure, P <sub>i</sub>	8.5915 1.326 <del>1.201.36</del> N/A	E.1.a Lid Baseplate Shell Supports	<del>Table 3.4.73.E.8.1.1 of Docket 72-1008</del> 3.I.8.1 of Docket 72-1008 Table 3.4.7
E1.b	Design external pressure, P <sub>o</sub>	15 1.326 <del>38.51.17</del>	E.1.b Lid Baseplate Shell	P <sub>i</sub> bounds P <sub>i</sub> bounds <i>Buckling (methodology in 3.H of Docket 72-1008)3.H (Case 4) (buckling) of Docket 72-1008</i>
E1.c	Design internal pressure, P <sub>i</sub> , plus Temperature T	N/A <del>1.09-1.4</del>	Supports E1.c Shell	Table 3.4.8
E2	D + H + (P <sub>i</sub> , P <sub>o</sub> )	6.5 1.088 <del>2.64*0.9673(stress), 45.51.17(buckling)</del> 5.854.58	Lid Baseplate Shell Supports	3.E.8.1.2 of Docket 72-1008 3.I.8.2 of Docket 72-1008 <del>Table 3.4.93-AA (stress) of Docket 72-1008</del> <i>Buckling (methodology in 3.H of Docket 72-1008)3.H (Case 4) (buckling) of Docket 72-1008</i> <del>Table 3.4.93-AA of Docket 72-1008</del>

*Note: 0.967 multiplier reflects increase in MPC shell design temperature to 500 deg. F*

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13

<sup>††</sup> Note that in analyses, bounding pressures are applied, i.e., in buckling calculations P<sub>o</sub> is used, and in stress evaluations either P<sub>o</sub> or P<sub>i</sub> is appropriate

**TABLE 3.4.4 (CONTINUED)**  
**MPC RESULTS - MINIMUM SAFETY FACTOR**

<b>Load Case I.D.</b>	<b>Load Combination<sup>†,††</sup></b>	<b>Safety Factor</b>	<b>Location in FSAR Where the Analysis is Performed</b>
E3 E3.a	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', end drop	2.8 1.28 1.7224	E.a Lid 3.E.8.2.1-2 of Docket 72-1008 Baseplate 3.I.8.3 of Docket 72-1008 Shell <i>Buckling (methodology in 3.H of Docket 72-10083.H (Case 5)</i>
E3.b	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', side drop 0 deg.	N/A	( <del>buckling</del> ) of Docket 72-1008 Supports
E3.c	(P <sub>i</sub> ,P <sub>o</sub> ) + D + H', side drop 45 deg.	2.8 1.28 1.064 1.18 1.82	E.b Lid end drop bounds Baseplate end drop bounds <i>Shell</i> Table 3.4.6 <i>Supports</i> Table 3.4.6 <del>Basket Supports: Appendix 3.Y</del>
		2.8 1.28 1.416 1.56	E.c. Lid end drop bounds Baseplate end drop bounds <i>Shell</i> Table 3.4.6 <i>Supports</i> Table 3.4.6

† The symbols used for the loadings are defined in Table 2.2.13

†† Note that in analyses, bounding pressures are applied, i.e., in buckling calculations P<sub>o</sub> is used, and in stress evaluations either P<sub>o</sub> or P<sub>i</sub> is appropriate

**TABLE 3.4.4 (CONTINUED)**  
**MPC RESULTS - MINIMUM SAFETY FACTOR**

Load Case I.D.	Load Combination <sup>†, ††</sup>	Safety Factor	Location in FSAR
E4	T	Subsection 3.4.4.2 shows there are no primary stresses from thermal expansion.	Subsection 3.4.4.2
E5	D + T* + (P <sub>i</sub> *, P <sub>o</sub> *)	27.2 1.78 <del>1.1508 (buckling)</del> 13.64.16 (stress) N/A	Lid 3.E.8.2.1.3 of Docket 72-1008 Baseplate 3.I.8.4 of Docket 72-1008 Shell <del>Buckling (methodology in 3.H of Docket 72-1008) 3.H (Case 6) (buckling) of Docket 72-1008</del> 3.4.4.3.1.5 (thermal stress) of Docket 72-1008 Supports N/A

† The symbols used for the loadings are defined in Table 2.2.13.

†† Note that in analyses, bounding pressures are applied, i.e., in buckling calculations P<sub>o</sub> is used, and in stress evaluations either P<sub>o</sub> or P<sub>i</sub> is appropriate.

**TABLE 3.4.5  
HI-STORM 100 STORAGE OVERPACK AND HI-TRAC RESULTS - MINIMUM SAFETY FACTORS**

Load Case I.D.	Loading <sup>†</sup>	Safety Factor	Location in FSAR
01	D + H + T + (P <sub>o</sub> , P <sub>i</sub> )	1.33	Overpack
		N/A	Shell (inlet vent)/Base <del>3.4.3.53-D</del> Top Lid N/A
02	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (end drop/tip-over)	2.83(125); 2.29(100)	HI-TRAC
		2.604 (ASME Code limit) 2.61 (ASME Code limit) 2.91; 1.11(optional bolts)	Shell <del>3-AB</del> ; 3.4.3.3; 3.4.3.4 Pool Lid <del>3-AB</del> 3.4.3.8 Top Lid <del>3-AB</del> N/A Pocket Trunnion <del>3.4.4.3.3.13-AA</del> ; <del>3-AI</del>
02.a	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (end drop/tip-over)	1.606	Overpack
02.b	D + H' + (P <sub>o</sub> , P <sub>i</sub> ) (side drop)	1.134	Shell <del>3-M</del> ; 3.4.4.3.2.3 Top Lid 3.4.4.3.2.2
03	D (water jacket)	2.09	HI-TRAC
04	M (small and medium penetrant missiles)	1.392	Shell <del>3-Z</del> ; 3.4.9
		1.651	Transfer Lid <del>3-AD</del> ; 3.4.4.3.3.3 Top Lid <del>3-AH</del> ; 3.4.4.3.3.5
		1.168	<del>3-AG</del> ; 3.4.4.3.3.4
		2.65 (Side Strike); 1.35(End strike)	Overpack 3.4.8.1
		1.23 (End Strike)	HI-TRAC 3.4.8.2.1

<sup>†</sup> The symbols used for the loadings are defined in Table 2.2.13.

**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 <sub>m</sub> )	3.46 € (1134)	4.83 (396)	3.01 (1603)	4.36 (1603)]
Fuel Basket - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	1.32 (1065)	1.33 (577)	2.18 (1590)	1.44 (774)
Enclosure Vessel - Primary Membrane (P <sub>m</sub> )	6.54*.967 (1354)	6.62*.967 (1370)	6.56*.967 (2393)	6.86*.967 (2377)
Enclosure Vessel - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	2.52*.967 (1278)	2.99*.967 (1247)	1.10*.967 (1925)	1.56*.967 (1925)
Basket Supports – Primary Membrane (P <sub>m</sub> )	N/A	N/A	7.15 (1710)	9.37 (1699)
Basket Supports - Local Membrane Plus Primary Bending (P <sub>L</sub> +P <sub>b</sub> )	N/A	N/A	1.18 (1715)	1.56 (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 in this Revision (Table 2.2.3). Deleted.

**TABLE 3.4.6 (CONTINUED)**  
**MINIMUM SAFETY FACTORS FOR MPC COMPONENTS DURING TIP-OVER**  
**45g DECELERATIONS**

Component - Stress Result	MPC-32	
	0 Degrees	45 Degrees
Fuel Basket - Primary Membrane ( $P_m$ )	3.51 (715)	4.96 (366)
Fuel Basket - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	1.51 (390)	1.28 (19)
Enclosure Vessel - Primary Membrane ( $P_m$ )	4.11*.967 (1091)	5.59*.967 (1222)
Enclosure Vessel - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	1.11*.967 (1031)	1.46*.967 (1288)
Basket Supports - Primary Membrane ( $P_m$ )	3.44 (905)	4.85 (905)
Basket Supports - Local Membrane Plus Primary Bending ( $P_L+P_b$ )	1.30 (901)	1.71 (908)

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 in this Revision (Table 2.2.3). ~~Deleted.~~

**TABLE 3.4.6 (CONTINUED)**  
**MINIMUM SAFETY FACTORS FOR MPC-24E COMPONENTS DURING TIP-OVER**  
**45g DECELERATIONS**

Components – Stress Result	0 Degrees	45 Degrees
Fuel Basket – Primary Membrane ( $P_m$ )	-10,050 (3.67)	-7,021 (5.26)
Fuel Basket – Primary Membrane plus Primary Bending ( $P_L + P_b$ )	31,912 (1.73)	30,436 (1.82)
Enclosure Vessel – Primary Membrane ( $P_m$ )	6,586 (6.59*.967)	6,534 (6.65*.967)
Enclosure Vessel – Primary Membrane plus Primary Bending ( $P_L + P_b$ )	23,100 (2.82*.967)	17,124 (3.80*.967)

- Notes: 1. All stresses are reported in psi units and are based on closed gaps (primary stresses only).  
2. —~~2.~~—The numbers shown in parentheses are the corresponding safety factors.  
3. *Multiplier of 0.967 reflects increase in Enclosure Vessel Design Temperature from 450 deg. F to 500 deg. F in this Revision (Table 2.2.3).*



**TABLE 3.4.7**  
**STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY -**  
**INTERNAL PRESSURE ONLY**

Locations (Per Fig. 3.4.11)	Calculated Value of Stress Intensity (psi)	Category	Table 3.1.13 Allowable Value (psi) <sup>†</sup>	Safety Factor (Allowable/Calculated)
<u>Top Lid</u>				
A	1,633,641	P <sub>L</sub> + P <sub>b</sub>	25,450,300	15.616
Neutral Axis	21,920.2	P <sub>m</sub>	16,950,500	774.866
B	1,604,605	P <sub>L</sub> + P <sub>b</sub>	25,450,300	15.916
C	695,687	P <sub>L</sub> + P <sub>b</sub>	25,450,300	36.638
Neutral Axis	732,731	P <sub>m</sub>	16,950,500	23.223
D	2,962,960	P <sub>L</sub> + P <sub>b</sub>	25,450,300	8.598
<u>Baseplate</u>				
E	19,773,683	P <sub>L</sub> + P <sub>b</sub>	28,100,000	1.421
Neutral Axis	415,412	P <sub>m</sub>	18,700,000	45.148
F	20,601,528	P <sub>L</sub> + P <sub>b</sub>	28,100,000	1.361
G	9,610,695	P <sub>L</sub> + P <sub>b</sub>	28,100,000	2.923
Neutral Axis	2,268,278	P <sub>m</sub>	18,700,000	8.258
H	8,279,340	P <sub>L</sub> + P <sub>b</sub>	28,100,000	3.393

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid top), and 400 degrees F (baseplate bottom), and 500 degrees F (canister).

**TABLE 3.4.7 (CONTINUED)**  
**STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY -**  
**INTERNAL PRESSURE ONLY**

Locations (Per Fig. 3.4.11)	Calculated Value of Stress Intensity (psi)	Category	Table 3.1.13 Allowable Value (psi) <sup>†</sup>	Safety Factor (Allowable/Calculated)
<u>Canister</u>				
I	<del>6,788,860</del>	$P_m$	17,500	<del>2.582.55</del>
Upper Bending Boundary Layer Region	<del>7,202,189</del>	$P_L + P_b + Q$	52,500	<del>7.297.30</del>
	<del>7,014,044</del>	<del><math>P_L + P_b</math></del>	26,300	<del>3.753.73</del>
Lower Bending Boundary Layer Region	<del>43,645,986</del>	$P_L + P_b + Q$	<del>52,500</del>	<del>1.201.36</del>
	<del>11,349,621</del>	<del><math>P_L + P_b</math></del>	<del>26,300</del>	<del>3.322.82</del>

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid top), and 400 degrees F (baseplate bottom), and 500 degrees F (canister).

**TABLE 3.4.8  
PRIMARY AND SECONDARY STRESS INTENSITY RESULTS FOR  
CONFINEMENT BOUNDARY - PRESSURE PLUS THERMAL LOADING**

<b>Locations (Per Fig. 3.4.11)</b>	<b>Calculated Value of Stress Intensity (psi)</b>	<b>Category</b>	<b>Allowable Stress Intensity (psi)<sup>†</sup></b>	<b>Safety Factor (Allowable/Calculated)</b>
<u>Top Lid</u>				
A	7,8661,630	$P_L + P_b + Q$	50,85052,500	6.4632.2
Neutral Axis	6,55322.5	$P_m + P_L$	25,45026,300	3.881,169.
B	3,4091,604.1	$P_L + P_b + Q$	50,85052,500	14.932.7
C	13,646696	$P_L + P_b + Q$	50,85052,500	3.7375.5
Neutral Axis	12,182731	$P_m + P_L$	25,45026,300	2.0936.0
D	11,1452,960	$P_L + P_b + Q$	50,85052,500	4.5617.7
<u>Baseplate</u>				
E	19,41719,798	$P_L + P_b + Q$	56,10060,000	2.893.0
Neutral Axis	223,1410.0	$P_m + P_L$	28,10030,000	12673.2
F	19,86020,622	$P_L + P_b + Q$	56,10060,000	2.822.9
G	4,8364,789.4	$P_m + P_L + Q$	56,10060,000	11.612.5
Neutral Axis	1,2011,131.8	$P_m + P_L$	28,10030,000	23.426.5
H	4,4734,139.4	$P_L + P_b + Q$	56,10060,000	12.514.5

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid), 400 degrees F (baseplate), and 500 degrees F (canister).

**TABLE 3.4.8 (CONTINUED)**  
**PRIMARY AND SECONDARY STRESS INTENSITY RESULTS FOR**  
**CONFINEMENT BOUNDARY - PRESSURE PLUS THERMAL LOADING**

Locations (Per Fig. 3.4.11)	Calculated Value of Stress Intensity (psi)	Category	Allowable Stress Intensity (psi) <sup>†</sup>	Safety Factor (Allowable/Calculated)
<u>Canister</u>				
I	6,7996,787.4	$P_m \neq P_L$	26,30030,000	3.874.4
Upper Bending Boundary Layer Region	12,8134,200.5 12,1851,729.3	$P_L + P_b + Q$ $P_m \neq P_L$	52,500 26,300	4.1012.5 2.1615.2
Lower Bending Boundary Layer Region	48,37843,484 12,02810,498	$P_L + P_b + Q$ $P_m \neq P_L$	52,50060,000 26,30030,000	1.091.4 2.192.9

<sup>†</sup> Allowable stress intensities are evaluated at 550 degrees F (lid), 400 degrees F (baseplate), and 500 degrees F (canister).

**TABLE 3.4.9  
SAFETY FACTORS FROM SUPPLEMENTARY CALCULATIONS**

Item	Loading	Safety Factor	FSAR Location Where Details are Provided
HI-STORM Top Lid Weld Shear	Tipover	3.22	<del>3.4.4.3.2.23-K</del>
HI-STORM Lid Bottom Plate	End Drop	9.908	<del>3.4.4.3.2.33-M; 3-X</del>
HI-STORM Lid Bottom Plate Welds	End Drop	2.695	<del>3.4.4.3.2.33-M</del>
Pedestal Shield Compression	End Drop	1.011	<del>3.4.4.3.2.33-M</del>
HI-STORM Inlet Vent Plate Bending Stress	End Drop	1.606	<del>3.4.4.3.2.33-M</del>
HI-STORM Lid Top Plate Bending	End Drop –100 100S	5.29 1.625	<del>3.4.4.3.2.33-M</del>
HI-TRAC Pocket Trunnion Weld	HI-TRAC Rotation	2.92	<del>3.4.4.3.3.13-AA</del>
HI-TRAC 100 Optional Bolts - Tension	HI-TRAC Rotation	1.11	<del>3.4.4.3.3.13-AI</del>
HI-STORM 100 Shell	Seismic Event	18.6	3.4.7
HI-TRAC Transfer Lid Door Lock Bolts	Side Drop	2.387	<del>3.4.4.3.3.33-AD</del>
HI-TRAC Transfer Lid Separation	Side Drop	1.329	<del>3.4.4.3.3.33-AD</del>
HI-STORM 100 Top Lid	Missile Impact	<del>1.291-35</del>	<del>3.4.8.13-G</del>
HI-STORM 100 Shell	Missile Impact	2.77	<del>3.4.8.13-G</del>
HI-TRAC Water Jacket –Enclosure Shell Bending	Pressure	1.17	<del>3.4.4.3.3.43-AG</del>
HI-TRAC Water Jacket – Enclosure Shell Bending	Pressure plus Handling	1.14	<del>Subsection 3.4.4.3.3.1</del>
HI-TRAC Water Jacket – Bottom Flange Bending	Pressure	1.39	3.4.4.3.3.4
HI-TRAC Water Jacket – Weld	Pressure	1.42	<del>3.4.4.3.3.43-AG</del>
Fuel Basket Support Plate Bending	Side Drop	1.91	<del>3.4.4.3.1.83-Y</del>
Fuel Basket Support Welds	Side Drop	2.09	<del>3.4.4.3.1.83-Y</del>
MPC Cover Plates in MPC Lid	Accident Condition Internal Pressure	<del>1.349</del>	<del>3.4.4.3.1.83-Y</del>
MPC Cover Plate Weld	Accident Condition Internal Pressure	<del>5.846-04</del>	<del>3.4.4.3.1.83-Y</del>
HI-STORM Storage Overpack	External Pressure	2.88	<del>3.4.4.5.23-AK</del>
HI-STORM Storage Overpack Circumferential Stress	Missile Strike	2.49	3.4.8.1; <del>3-B</del>
HI-TRAC Transfer Cask Circumferential Stress	Missile Strike	2.61	3.4.8.2; <del>3-AM</del>
HI-TRAC Transfer Cask Axial Membrane Stress	Side Drop	2.09	<del>3-Z</del> ; 3.4.9.1

**TABLE 3.4.10**  
**INPUT DATA FOR SEISMIC ANALYSIS OF ANCHORED HI-STORM 100 SYSTEM**

Item	Data Used	Actual Value and Reference
Cask height, inch	231.25	231.25" (Dwg. 1495)
Contact diameter at ISFSI pad, inch	146.5	146.5 (Dwg. 3187)
Overpack empty, wt. Kips	270	267.87 (Table 3.2.1)
Bounding wt. of loaded MPC, kips	90	88.135 (Table 3.2.1)
Overpack-to-MPC radial gap (inch)	2.0	2.0' (Dwg. 1495, Sheets 2 and 5)
Overpack C.G. height above ISFSI pad, inch	117.0	116.8 (Table 3.2.3)
Overpack with Loaded MPC - C.G. height above ISFSI pad	118.5	118.5 (Table 3.2.3)
Applicable Response Spectra	Fig. 3.4-31 to 3.4-36	Figures 3.4-30
ZPA:	RG 1.60                      Western Plant	Site-Specific
Horizontal 1	1.5                                  1.45	
Horizontal 2	1.5                                  1.45	
Vertical	1.5                                  1.3	
No. of Anchor Studs	28	Up to 28
Anchor Stud Diameter		
Inch	2.0	2.0 (BOM 3189)
Yield stress, ksi	80 (minimum)	Table 1.2.7
Ultimate stress, ksi	125 (minimum)	Table 1.2.7
Free length, inch*	16-42	Site-specific
Pre-load tensile stress, ksi*	55-65	55-65

\*For the confirmatory dynamic analyses, bolt spring rates were computed using the maximum length, and the preload stress was slightly above 60.1 ksi. For the static analysis, all combinations were evaluated.

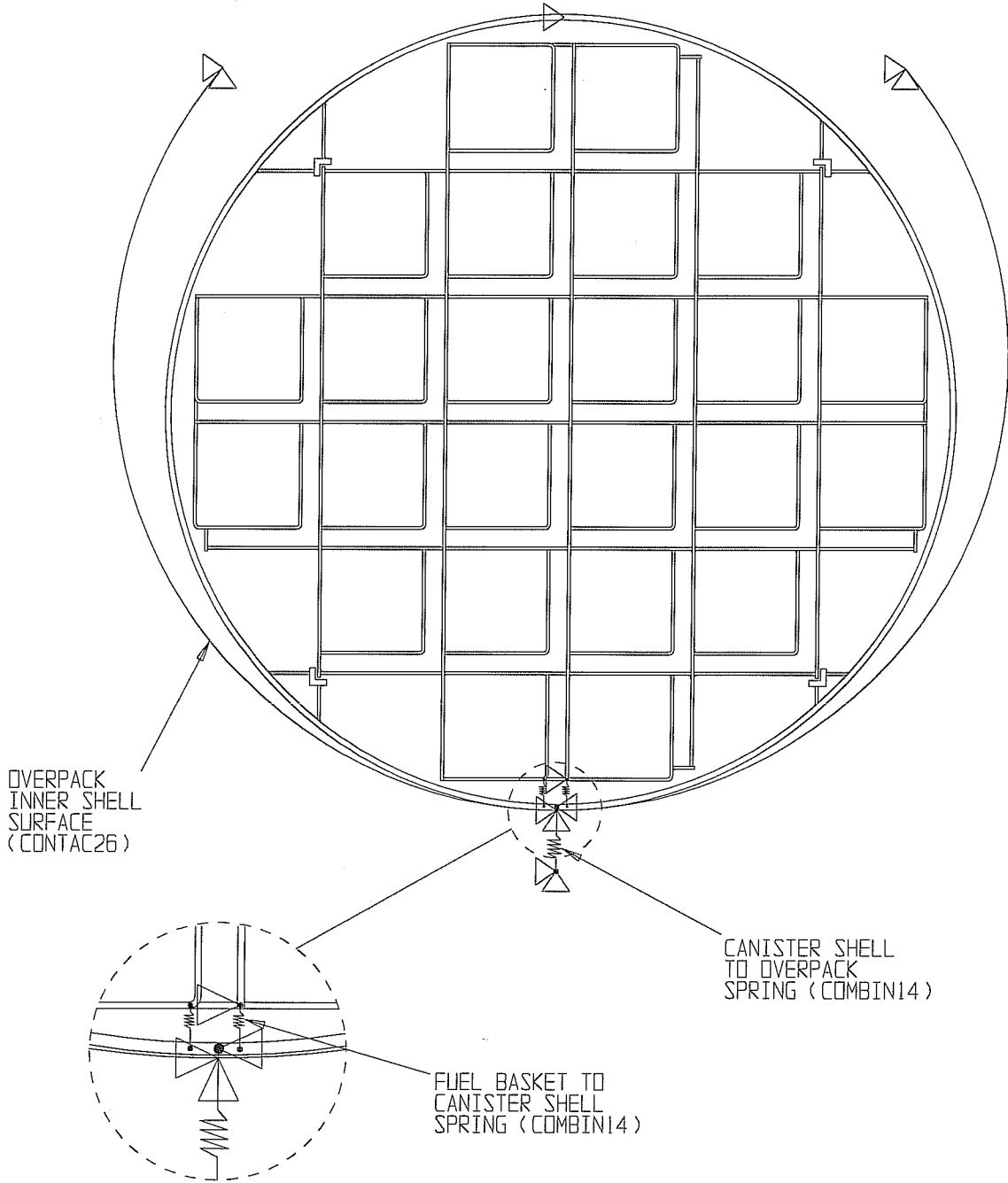
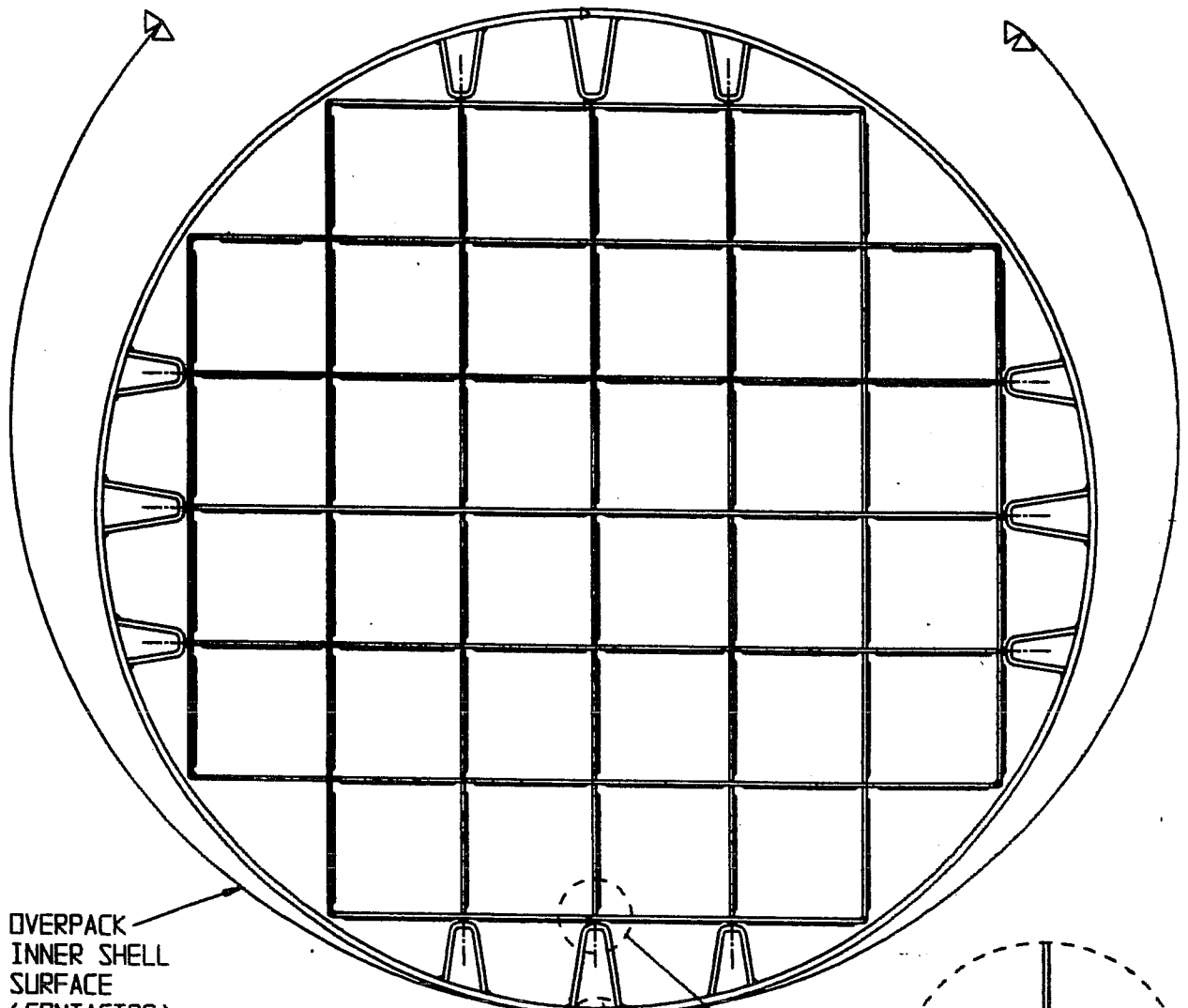


FIGURE 3.4.1; FINITE ELEMENT MODEL OF MPC-24



OVERPACK  
INNER SHELL  
SURFACE  
(CONTACT26)

CANISTER SHELL TO  
OVERPACK SPRING  
(COMBIN14)

FUEL BASKET TO  
CANISTER SHELL  
SPRING (COMBIN14)

△ = DISPLACEMENT  
CONSTRAINTS

FIGURE 3.4.2; FINITE ELEMENT MODEL OF MPC-32

(0 DEGREE DROP MODEL)



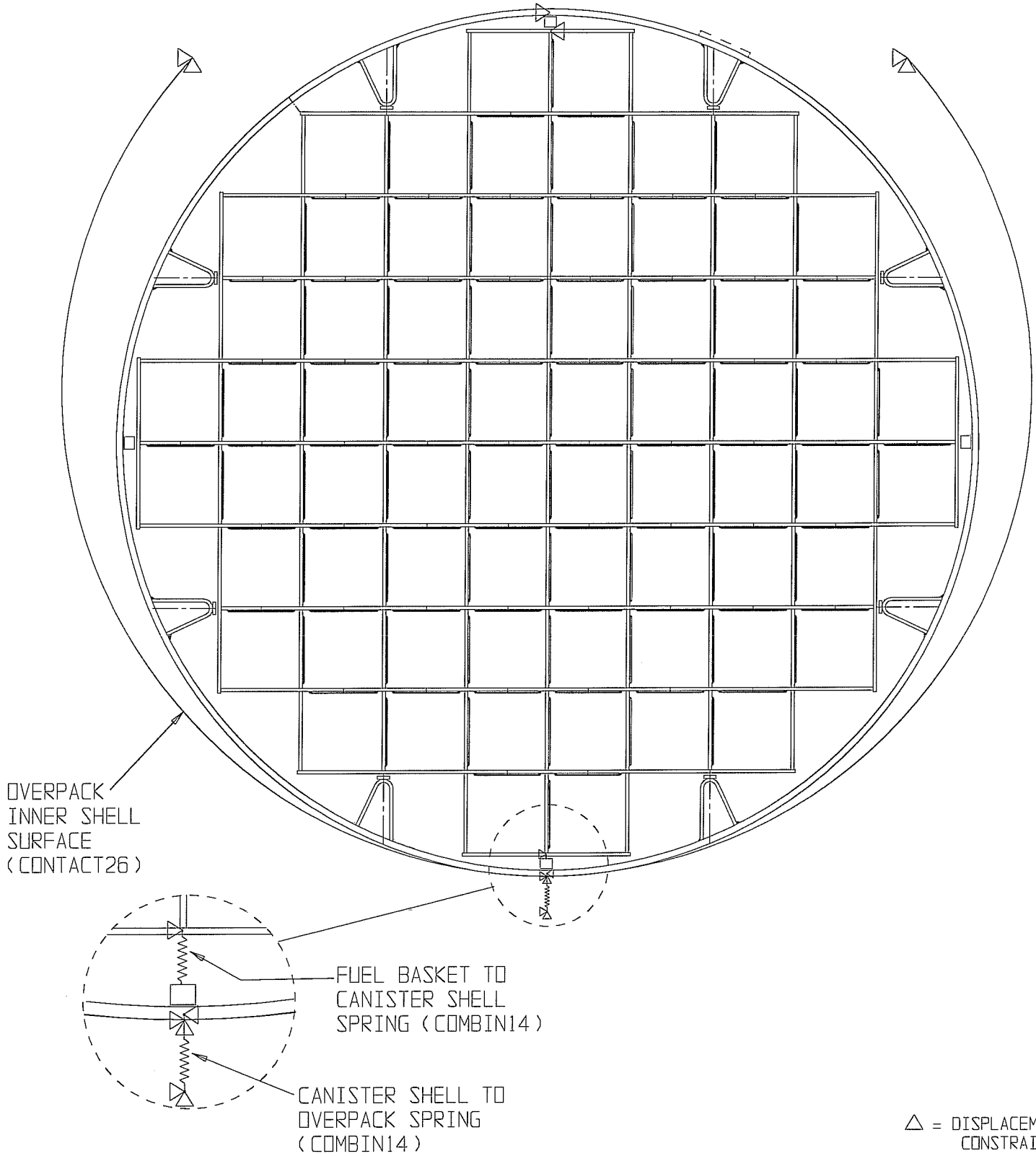


FIGURE 3.4.3; FINITE ELEMENT MODEL OF MPC-68

(0 DEGREE DROP MODEL)

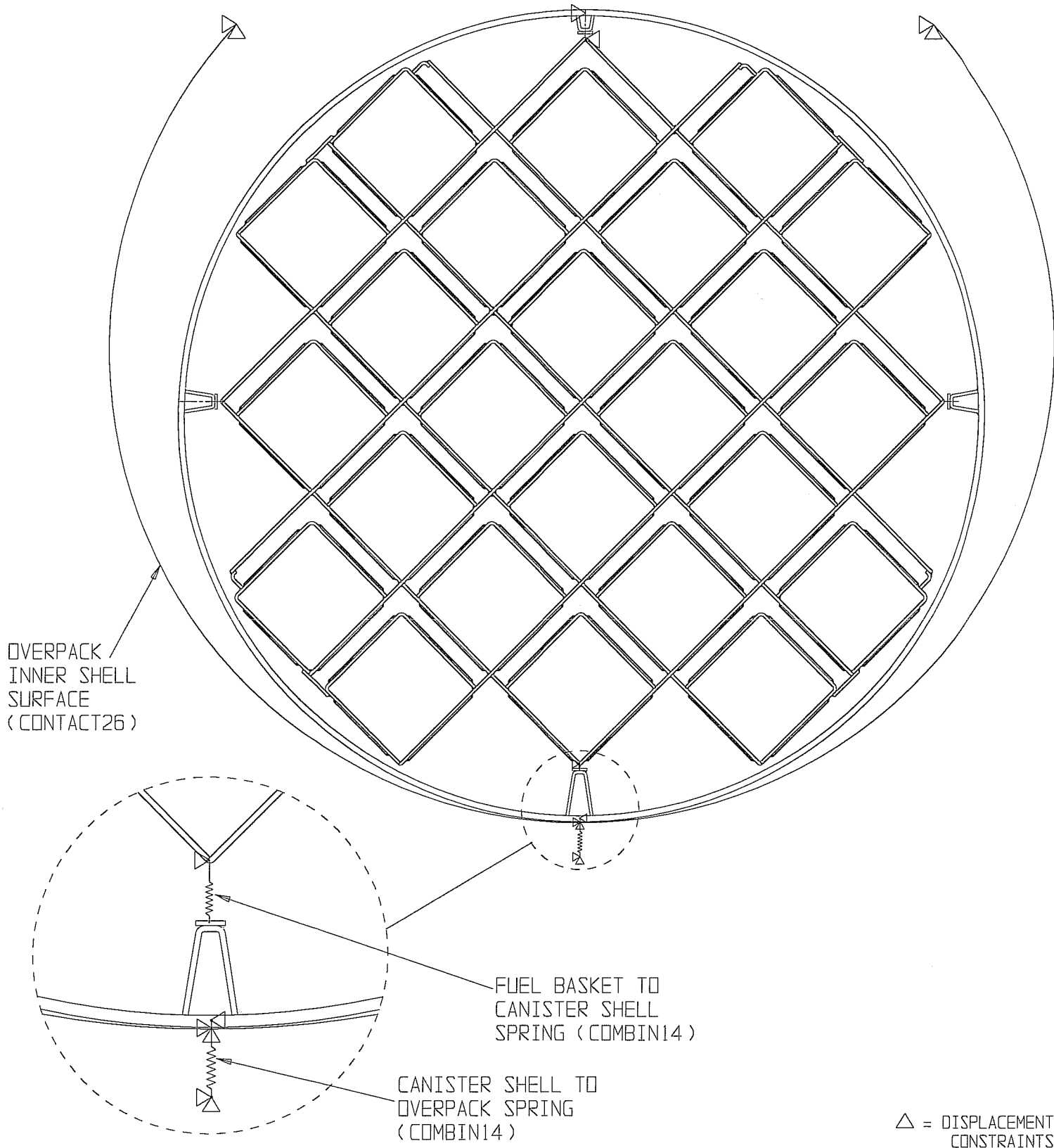


FIGURE 3.4.4; FINITE ELEMENT MODEL OF MPC-24

(45 DEGREE DROP MODEL)

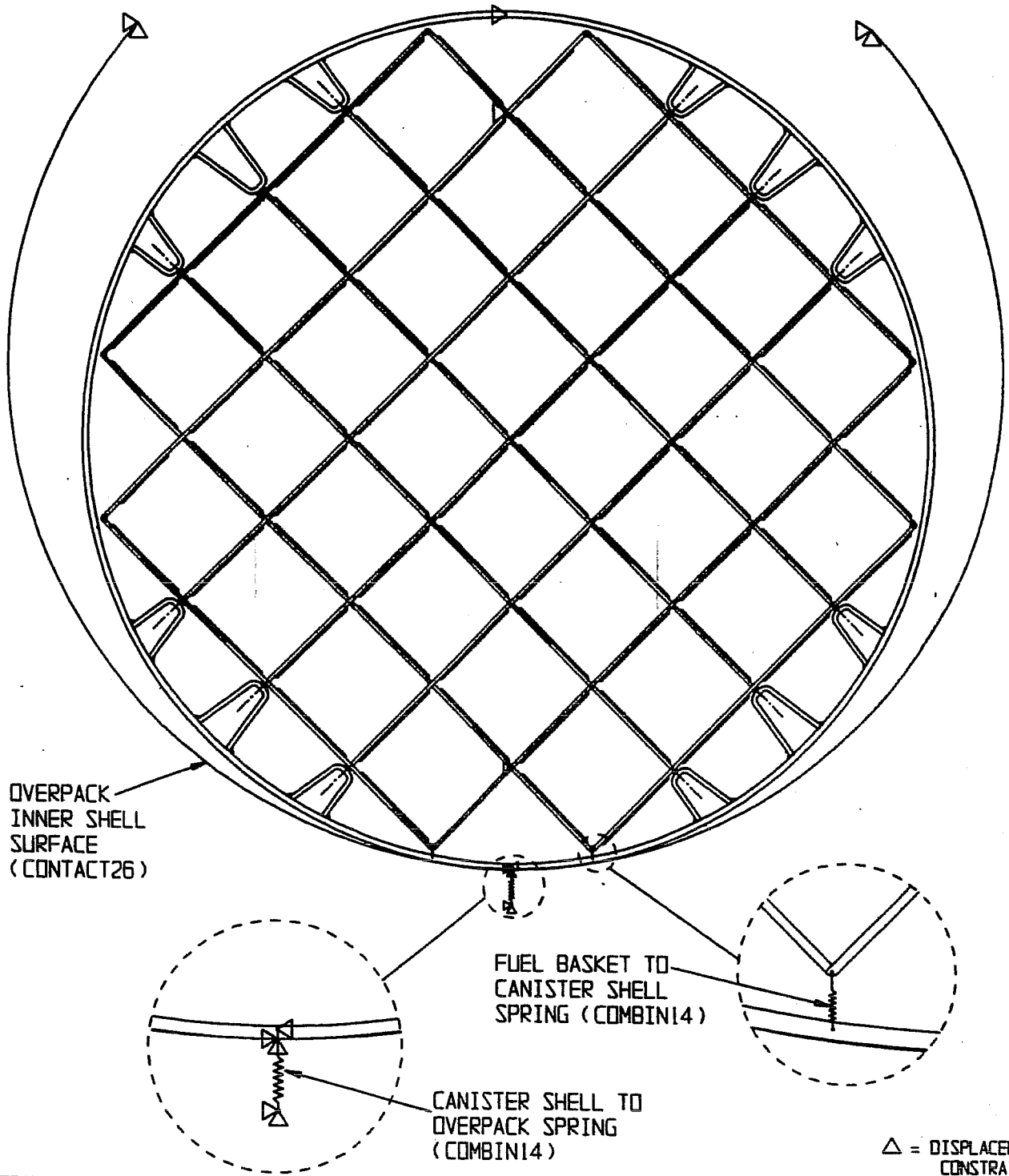


FIGURE 3.4.5; FINITE ELEMENT MODEL OF MPC-32

(45 DEGREE DRIP MODEL)

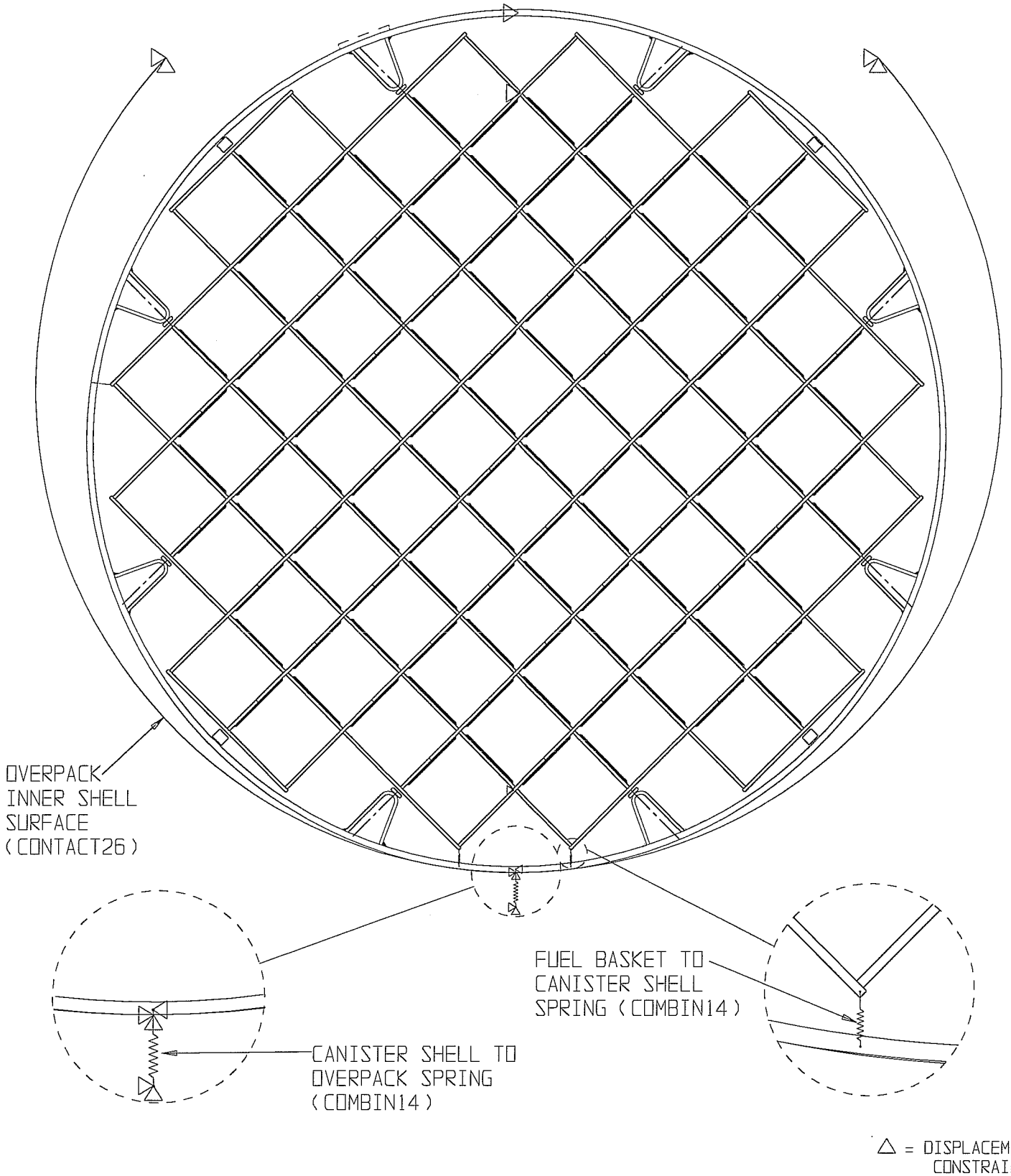
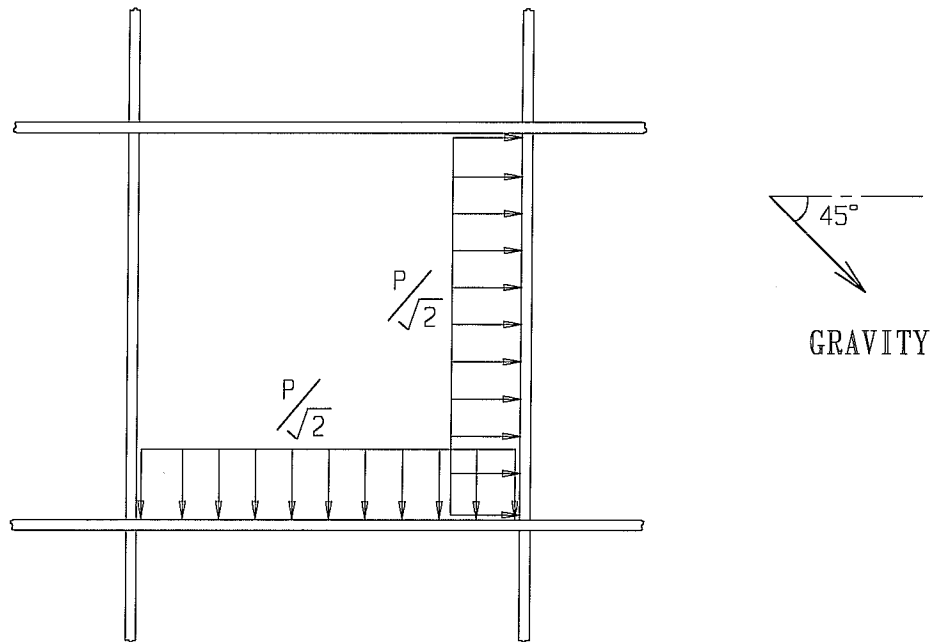
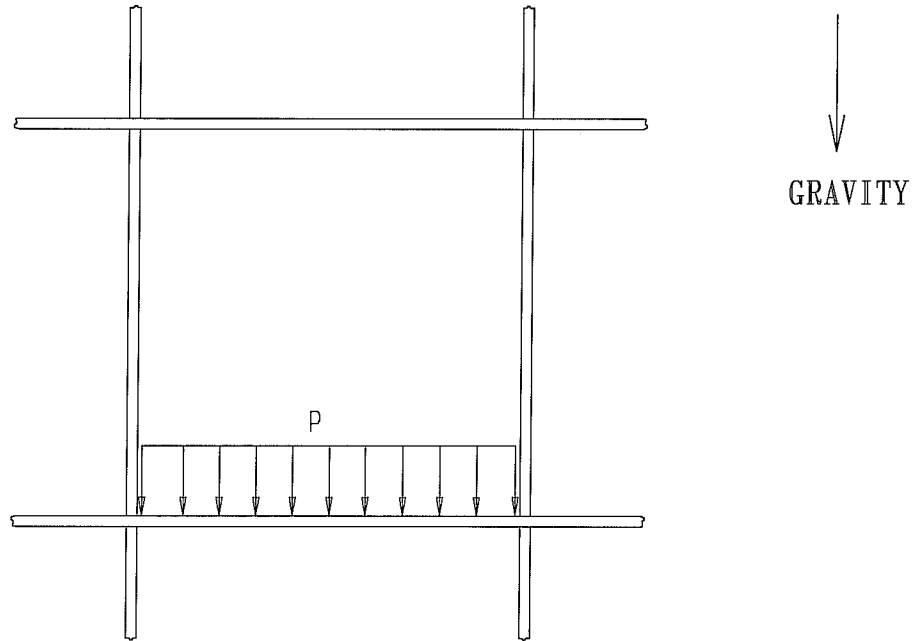
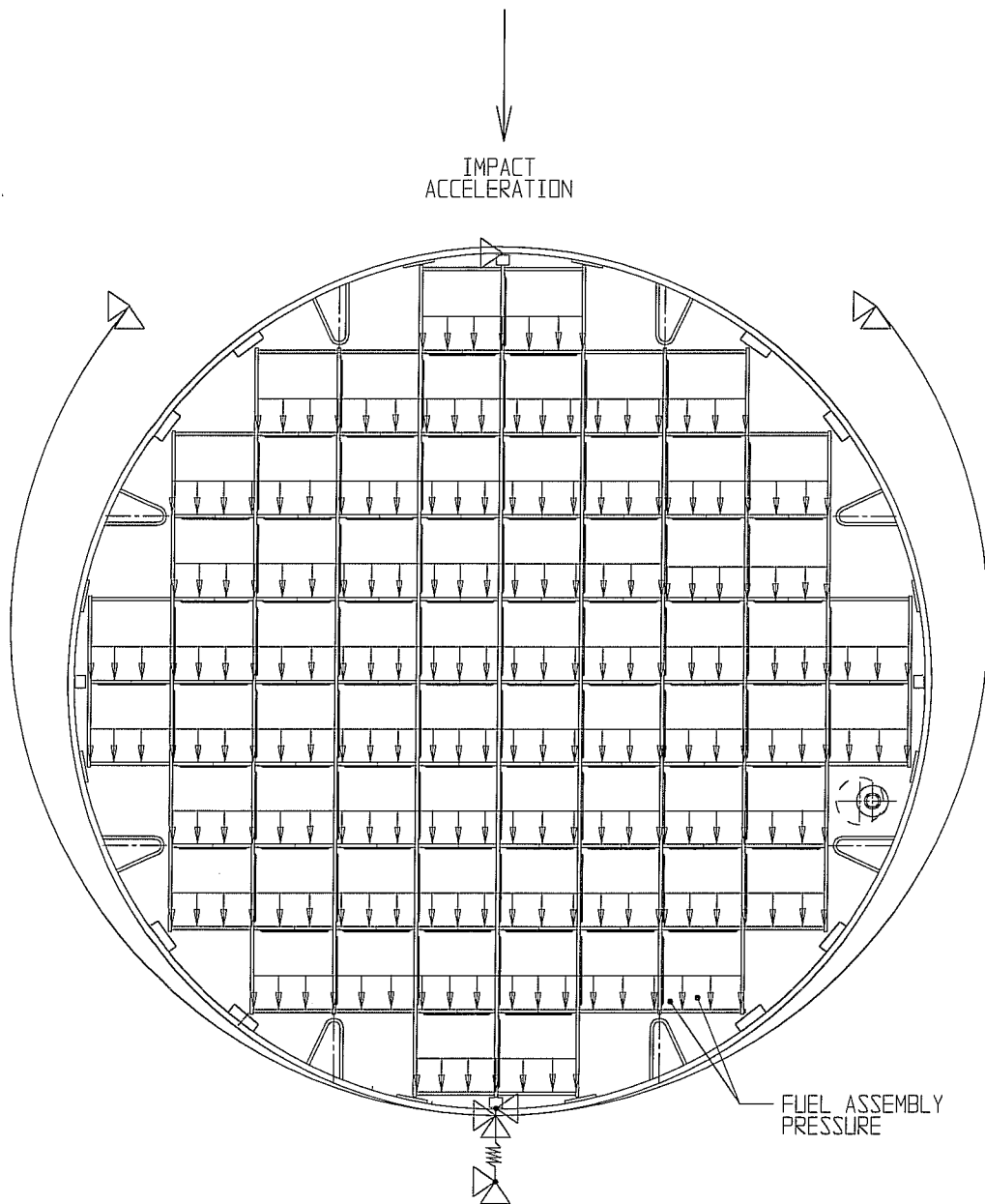


FIGURE 3.4.6; FINITE ELEMENT MODEL OF MPC-68

(45 DEGREE DROP MODEL)



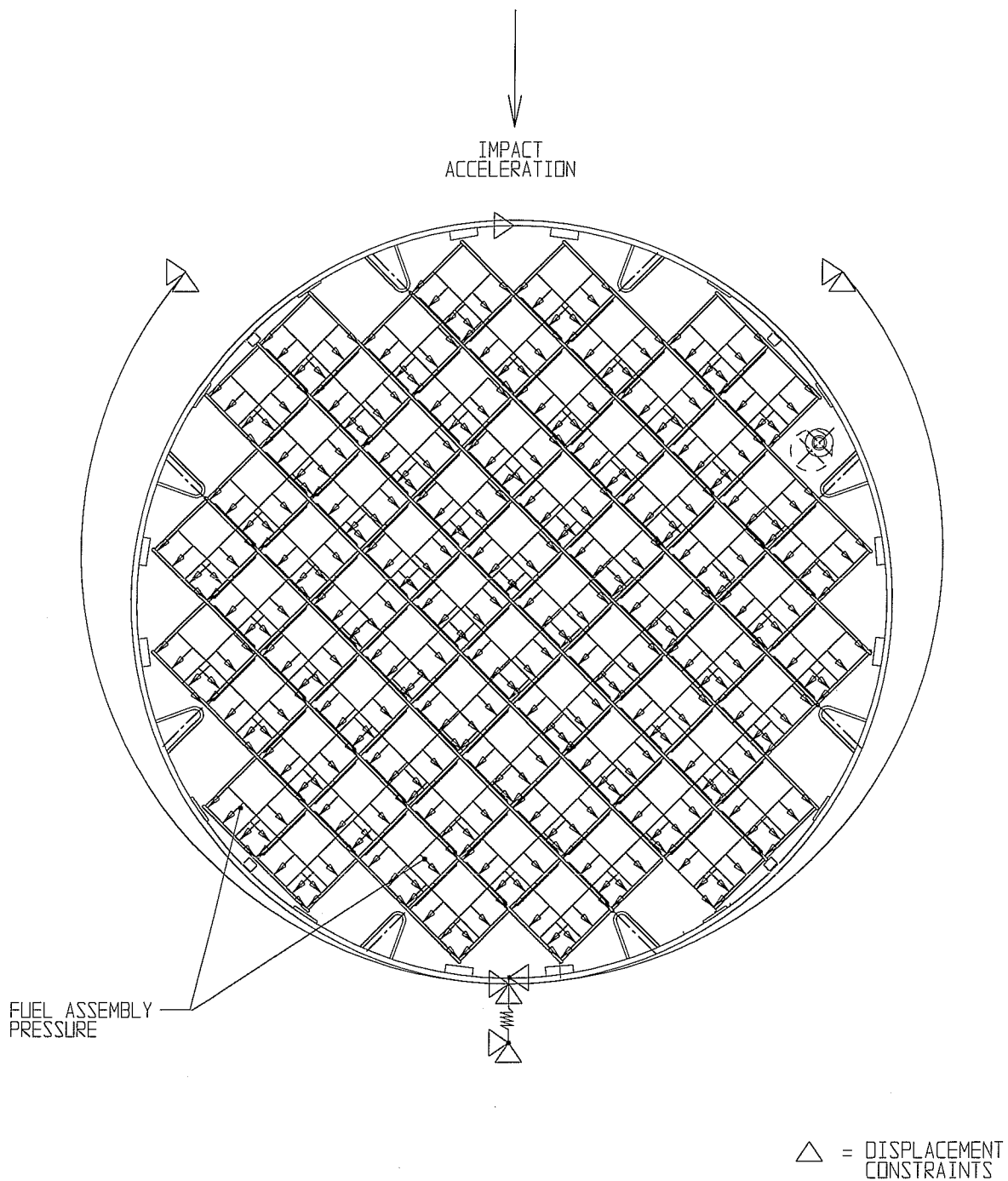
*FIGURE 3.4.7; DETAIL OF FUEL ASSEMBLY PRESSURE  
LOAD ON MPC BASKET*



FUEL ASSEMBLY PRESSURE

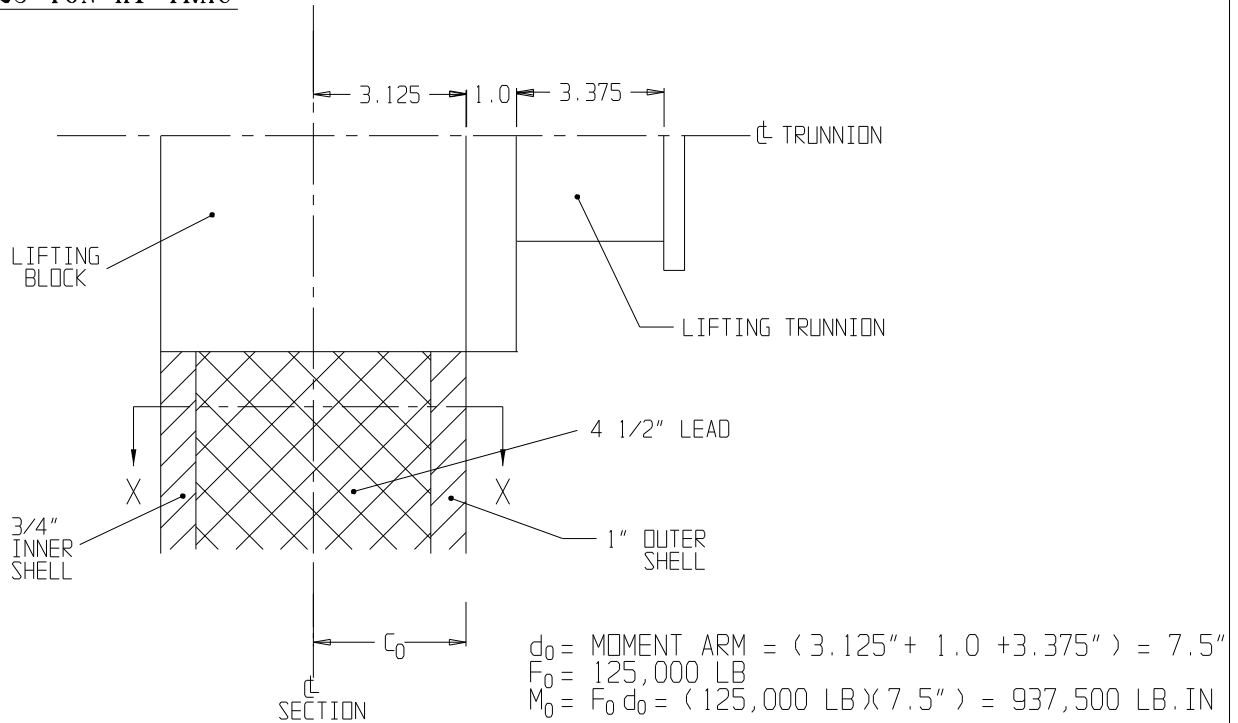
△ = DISPLACEMENT CONSTRAINTS

FIGURE 3.4.8; 0 DEGREE SIDE DROP OF MPC

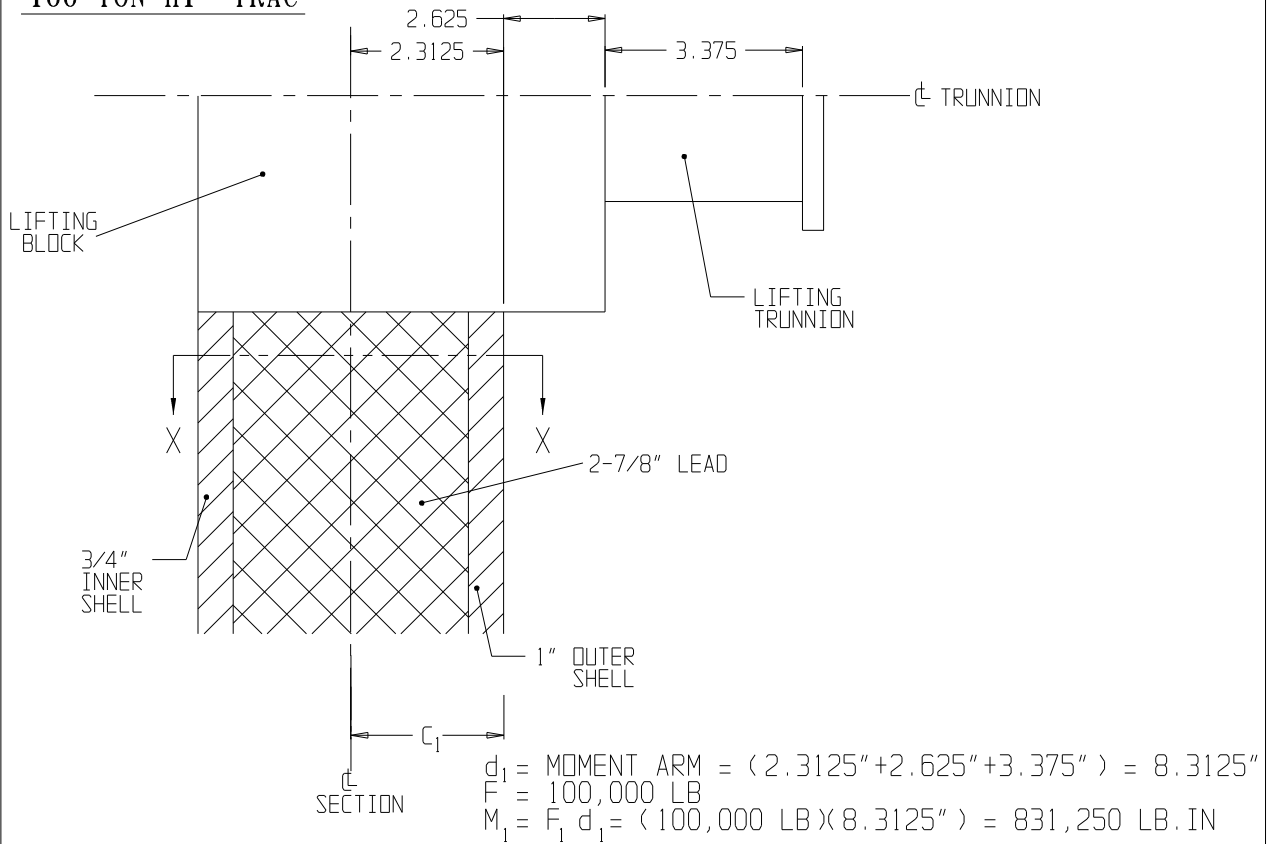


*FIGURE 3.4.9; 45 DEGREE SIDE DROP OF MPC*

## 125 TON HI-TRAC



## 100 TON HI-TRAC



**FIGURE 3.4.10; COMPARISON OF 125 TON AND 100 TON HI-TRAC LIFTING TRUNNION CONNECTION**

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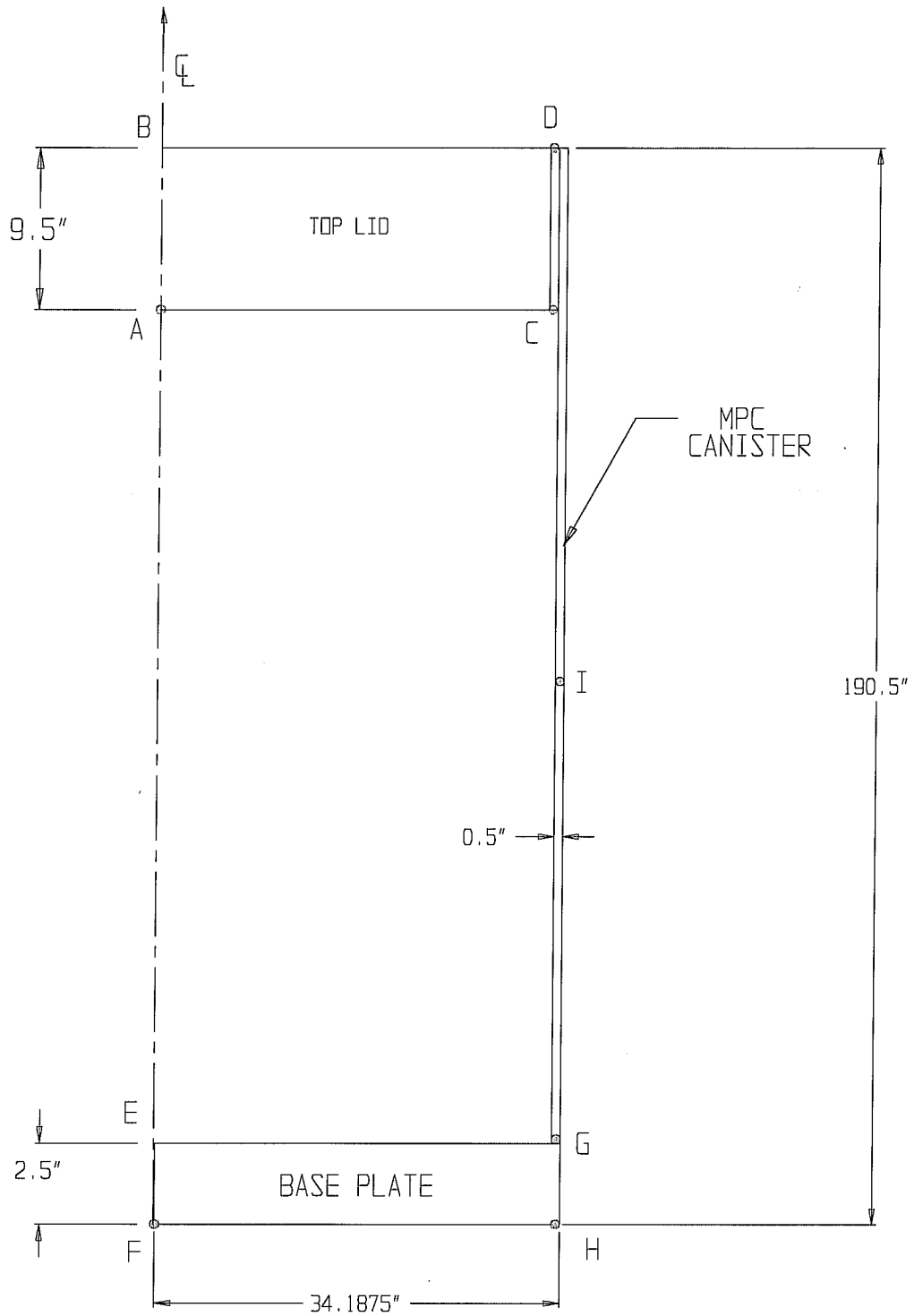


FIGURE 3.4.11 CONFINEMENT BOUNDARY MODEL SHOWING TEMPERATURE DATA POINTS

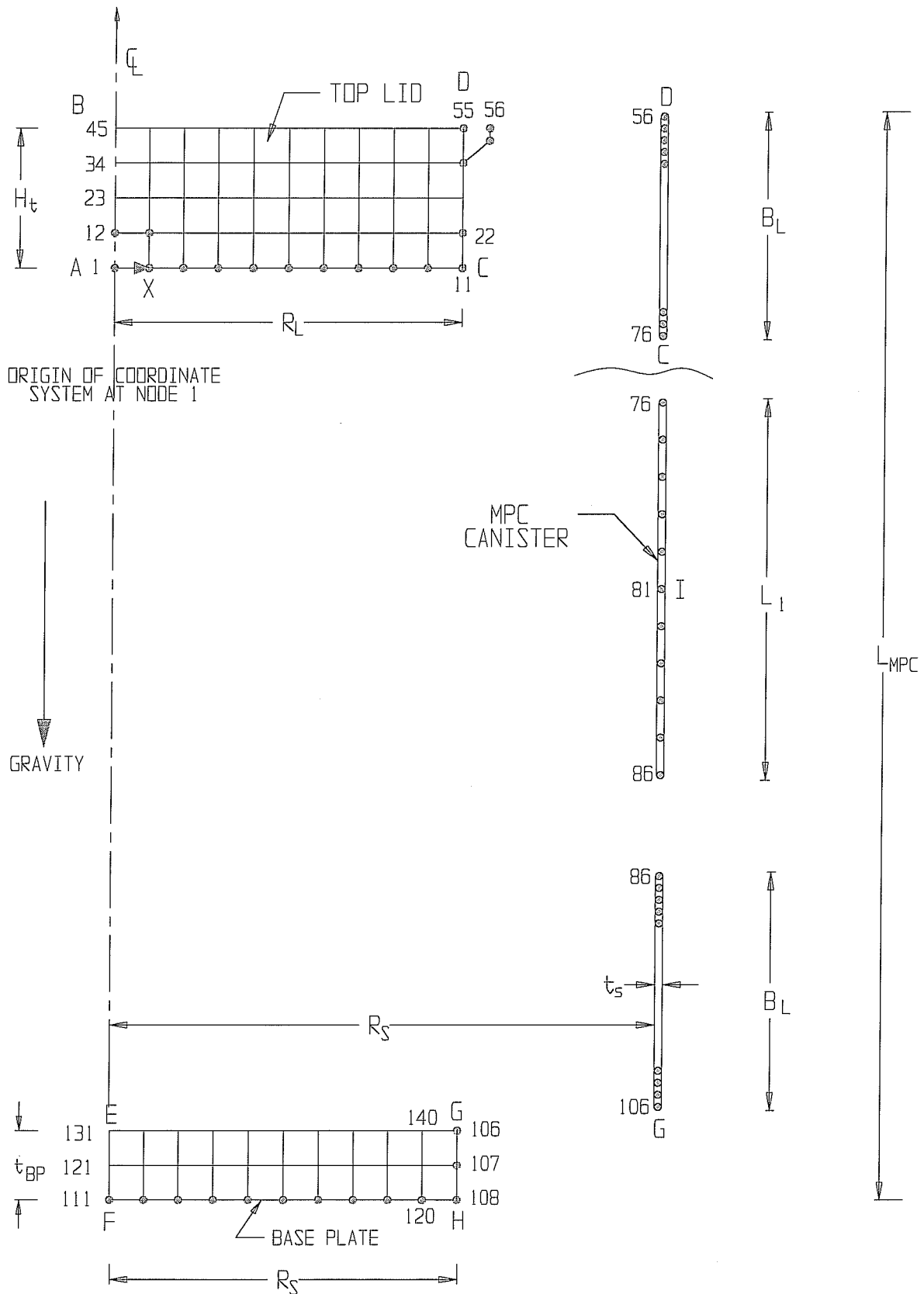
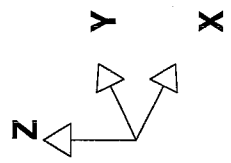
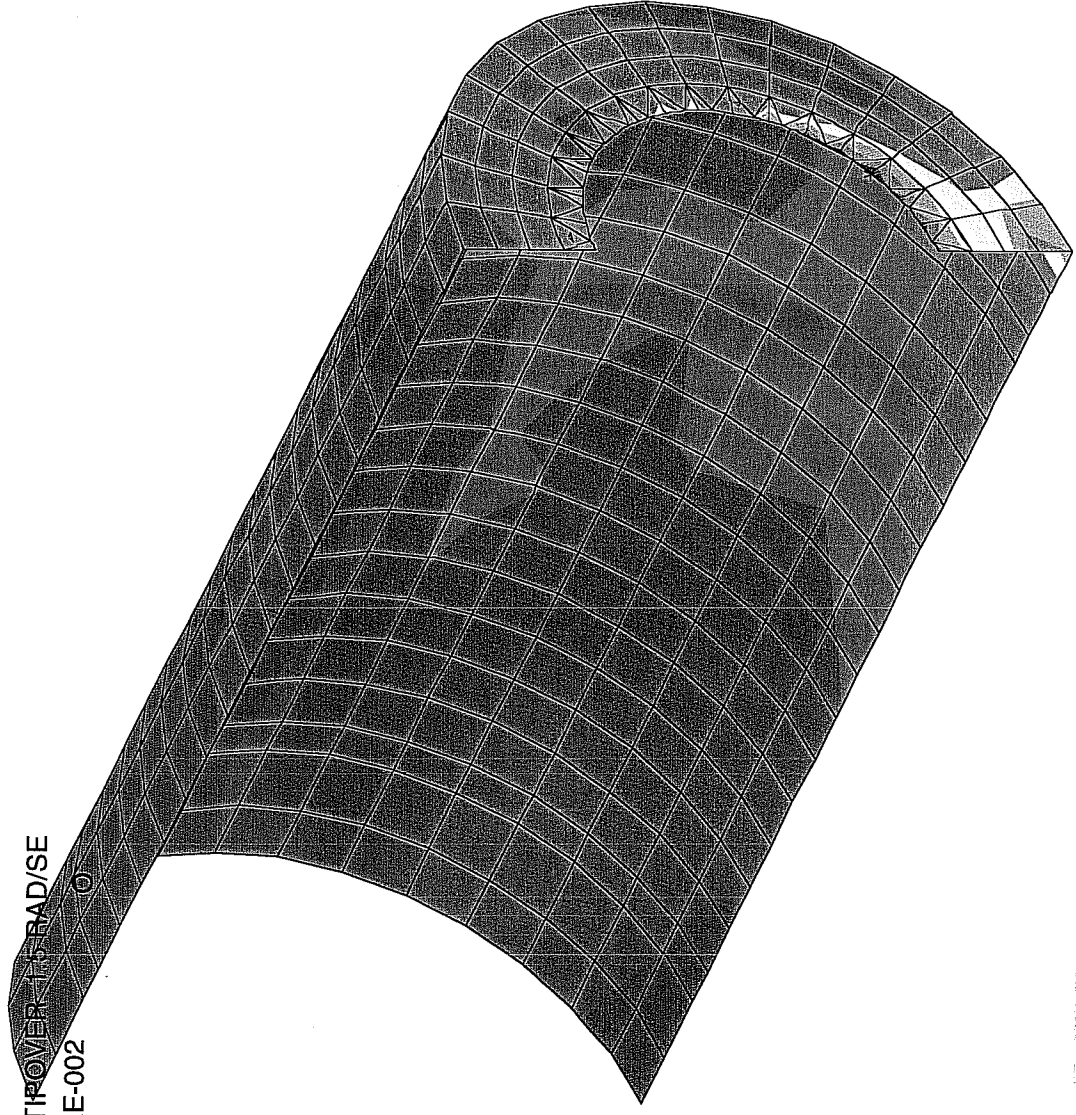


FIGURE 3.4.12 MPC - CONFINEMENT BOUNDARY FINITE ELEMENT GRID (EXPLODED VIEW)

HISTORM DEFORMABLE TIROVER TIRAD/SE  
 STEP 80 TIME = 7.9997852E-002  
 MAX\_VONMISES



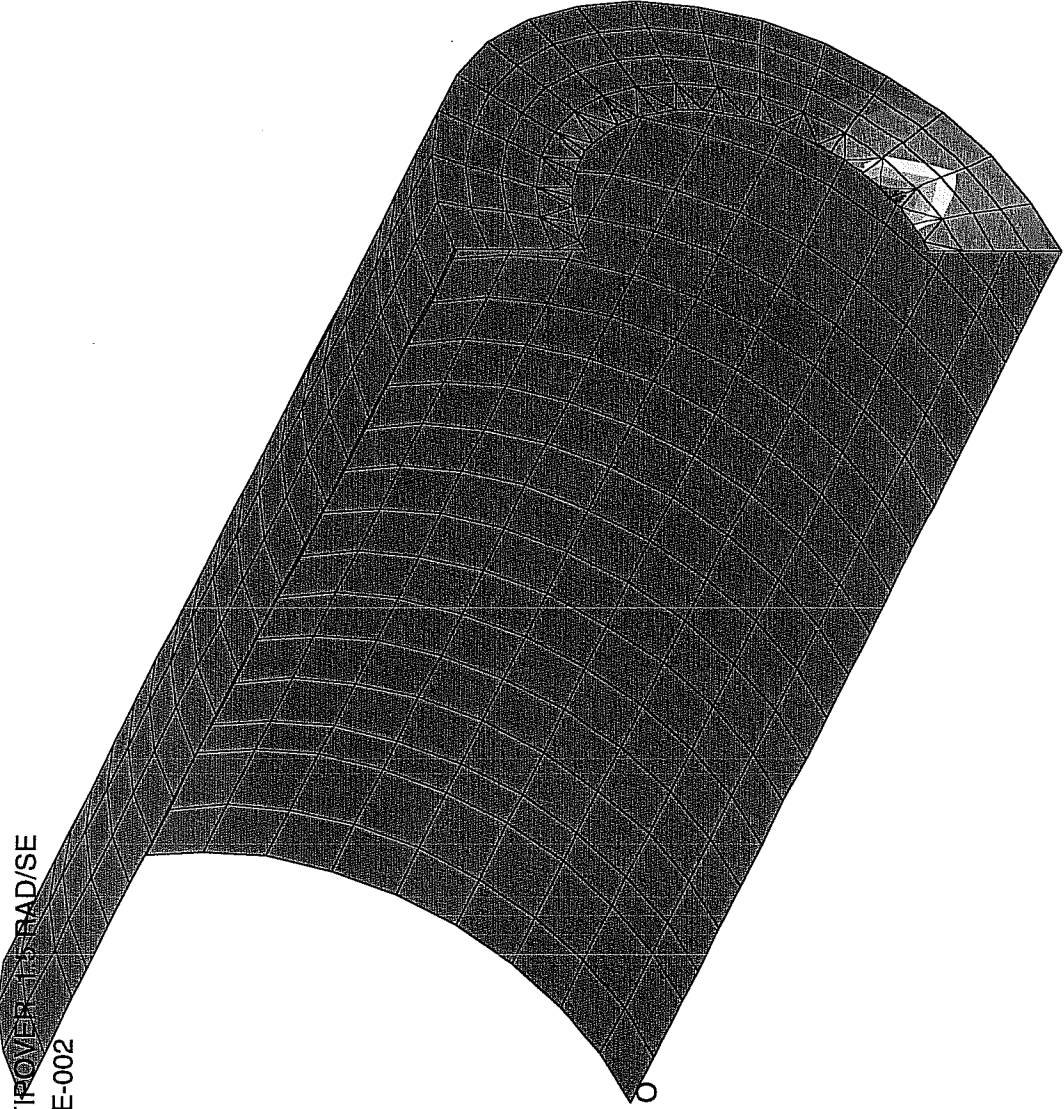
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- 3.3501E+004
- 3.0450E+004
- 2.7399E+004
- 2.4348E+004
- 2.1297E+004
- 1.8246E+004
- 1.5195E+004
- 1.2144E+004
- 9.0930E+003
- 6.0420E+003
- 2.9909E+003
- 2.9909E+003

HI-STORM FSAR  
 HI-2002444

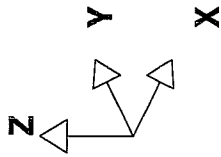
Fig. 3.4.13 Von Mises Stress Outer Shell

REV. 0

HISTORM DEFORMABLE TIRCOVER TIRAD/SE  
 STEP 80 TIME = 7.9997852E-002  
 PSTN(TOP)



- 2.7826E-002 \*
- 2.7826E-002
- 2.5043E-002
- 2.2261E-002
- 1.9478E-002
- 1.6696E-002
- 1.3913E-002
- 1.1130E-002
- 8.3478E-003
- 5.5652E-003
- 2.7826E-003
- 0.0000E+000
- 0.0000E+000



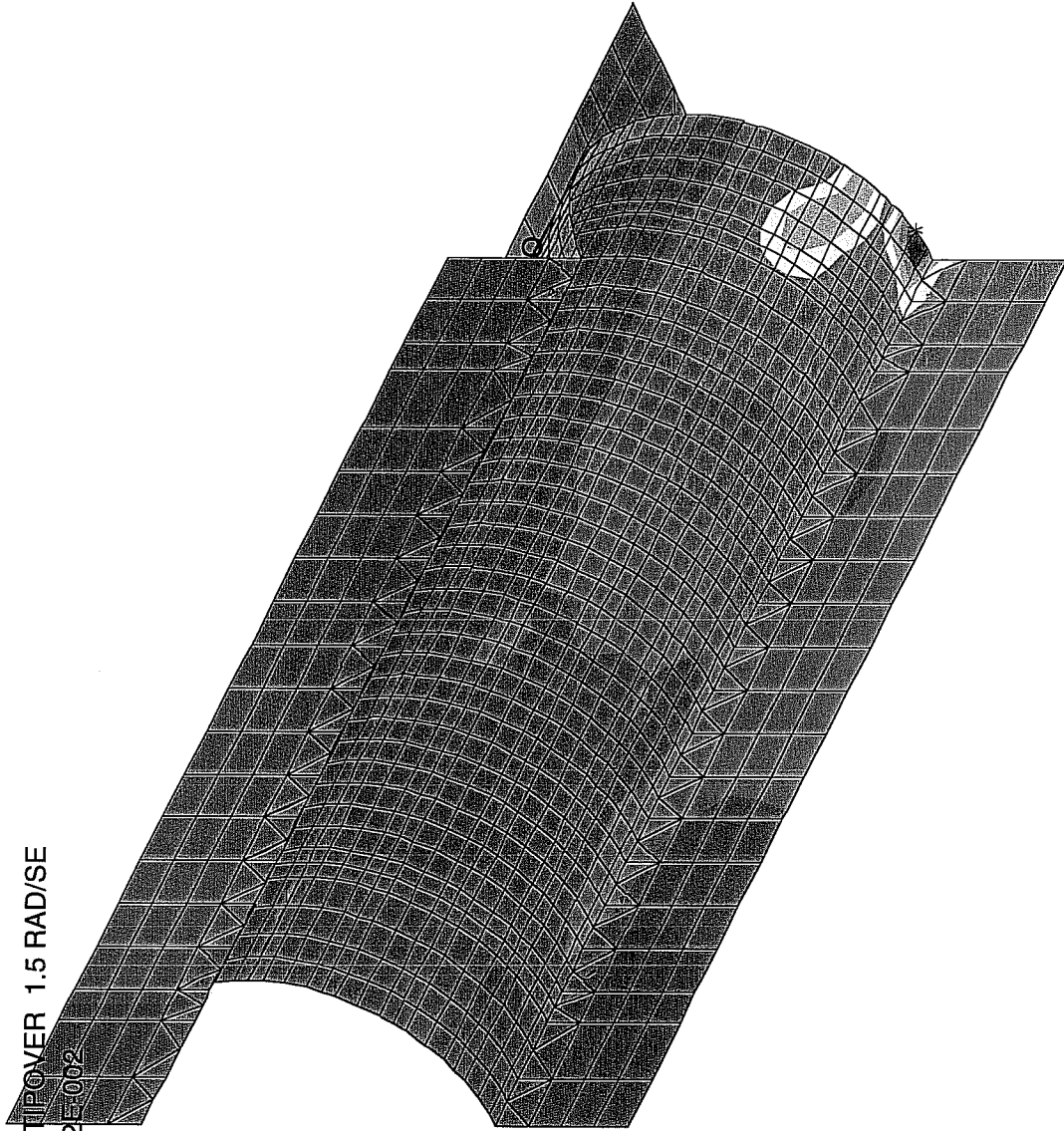
HI-2002444

HI-STORM FSAR

Fig. 3.4.14 Plastic Strain Outer Shell

Rev. 0

HISTORM DEFORMABLE TIP OVER 1.5 RAD/SE  
 STEP 80 TIME = 7.9997852E+002  
 MAX\_VONMISES



- 3.2444E+004 \*
- 3.2444E+004
- 2.9301E+004
- 2.6159E+004
- 2.3016E+004
- 1.9873E+004
- 1.6731E+004
- 1.3588E+004
- 1.0446E+004
- 7.3033E+003
- 4.1608E+003
- 1.0182E+003
- 1.0182E+003

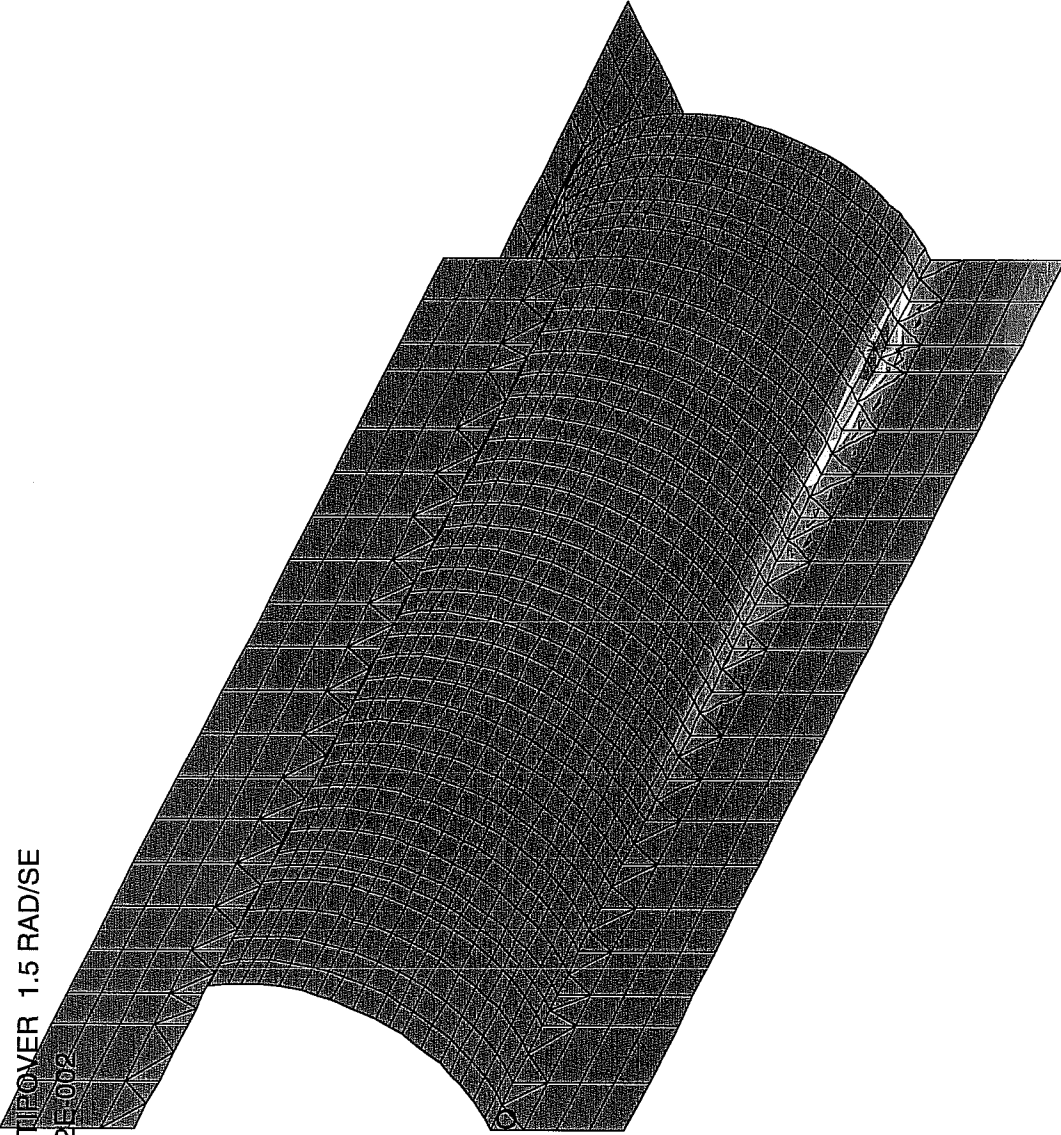
HI-2002444

Fig. 3.4.15 Von Mises Stress - Inner Shell

REV. 0

HI-STORM FSAR

HISTORM DEFORMABLE TUBES OVER 1.5 RAD/SE  
 STEP 80 TIME = 7.9997852E+002  
 PSTN(TOP)



- 4.5231E-003
- 4.5231E-003
- 4.0708E-003
- 3.6185E-003
- 3.1662E-003
- 2.7139E-003
- 2.2616E-003
- 1.8093E-003
- 1.3569E-003
- 9.0463E-004
- 4.5231E-004
- 0.0000E+000
- 0.0000E+000

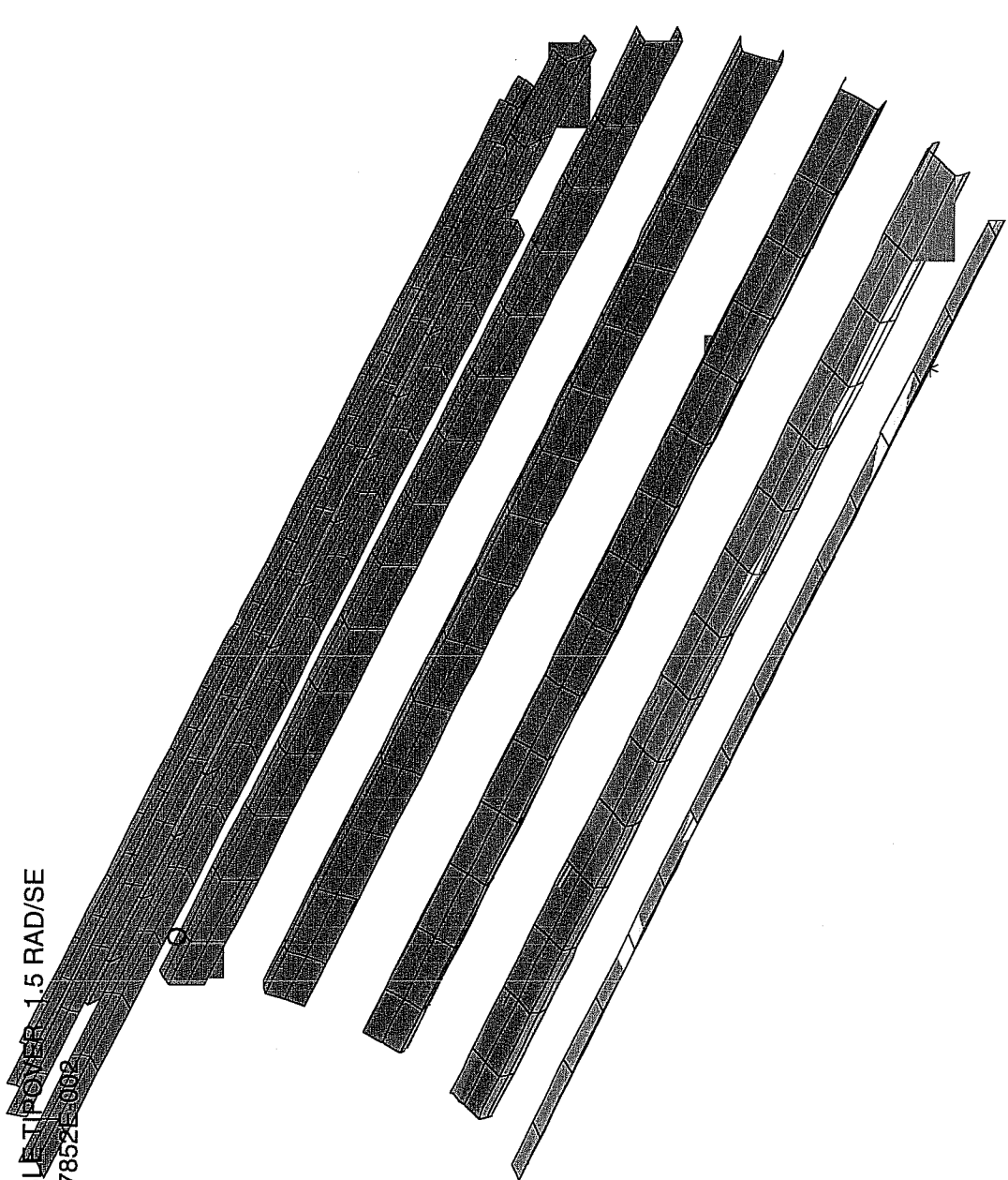
HI-2002444

HI-STORM FSAR

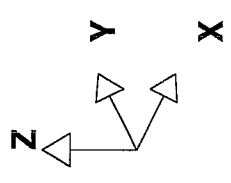
Fig. 3.4.16 Plastic Strain Inner Shell

Rev. 0

\*  
 3.4447E+004  
 3.4447E+004  
 3.1046E+004  
 2.7645E+004  
 2.4244E+004  
 2.0843E+004  
 1.7442E+004  
 1.4041E+004  
 1.0640E+004  
 7.2385E+003  
 3.8375E+003  
 4.3639E+002  
 4.3639E+002



HISTORM DEFORMABLE TIP POWER 1.5 RAD/SE  
 STEP 80 TIME = 7.9997852E+002  
 MAX\_VONMISES

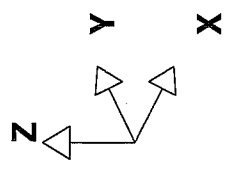
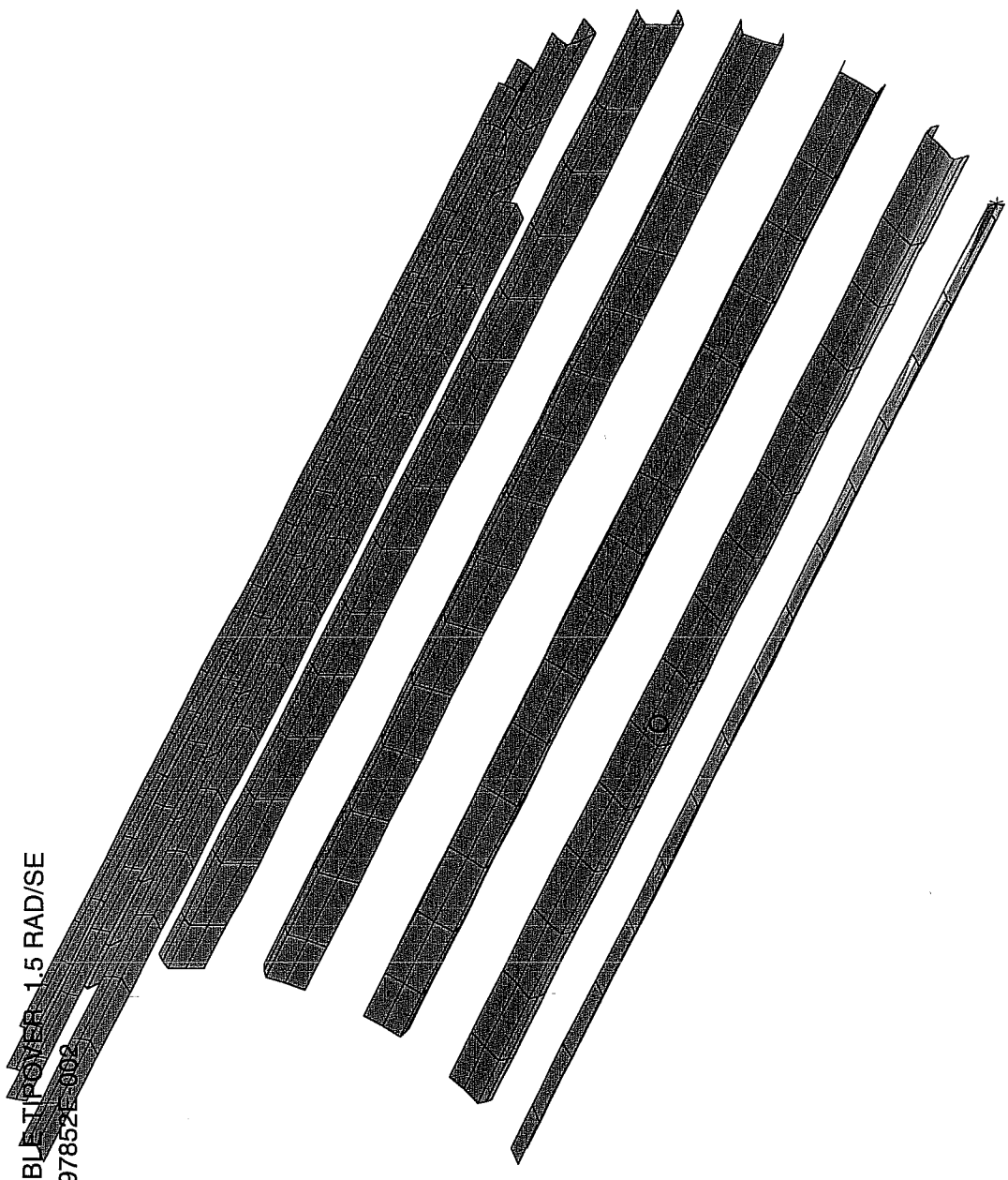


HI-2002444  
 HI-STORM FSAR

Fig. 3.4.16a Von Mises Stress - Channel

REV. 0

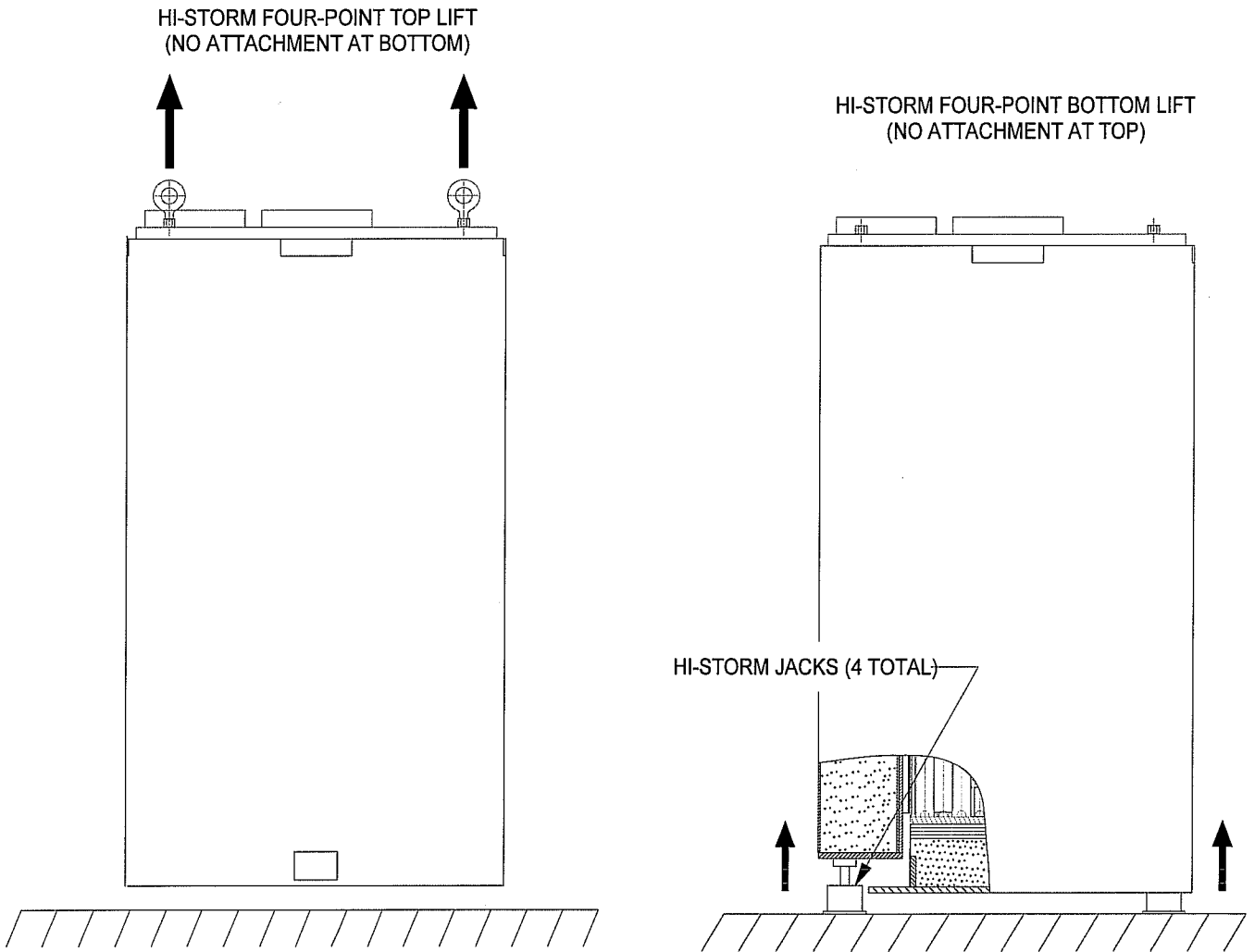
HISTORM DEFORMABLE T/POWER 1.5 RAD/SE  
 STEP 80 TIME = 7.9997852E-002  
 PSTN(MID)



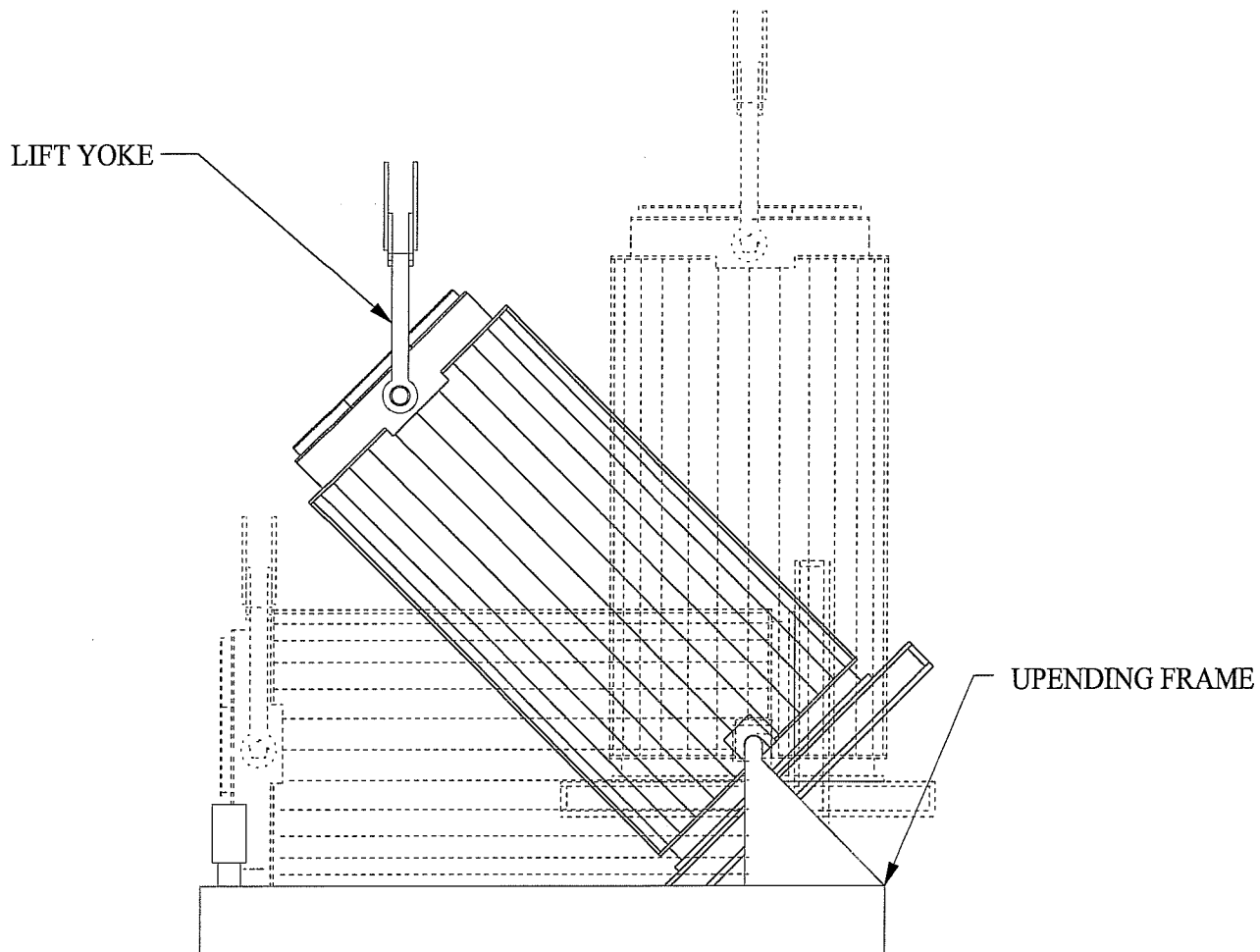
- 1.6285E-001
- 1.6285E-001
- 1.4656E-001
- 1.3028E-001
- 1.1399E-001
- 9.7707E-002
- 8.1423E-002
- 6.5138E-002
- 4.8854E-002
- 3.2569E-002
- 1.6285E-002
- 0.0000E+000
- 0.0000E+000

Fig. 3.4.16b Plastic Strain - Channel

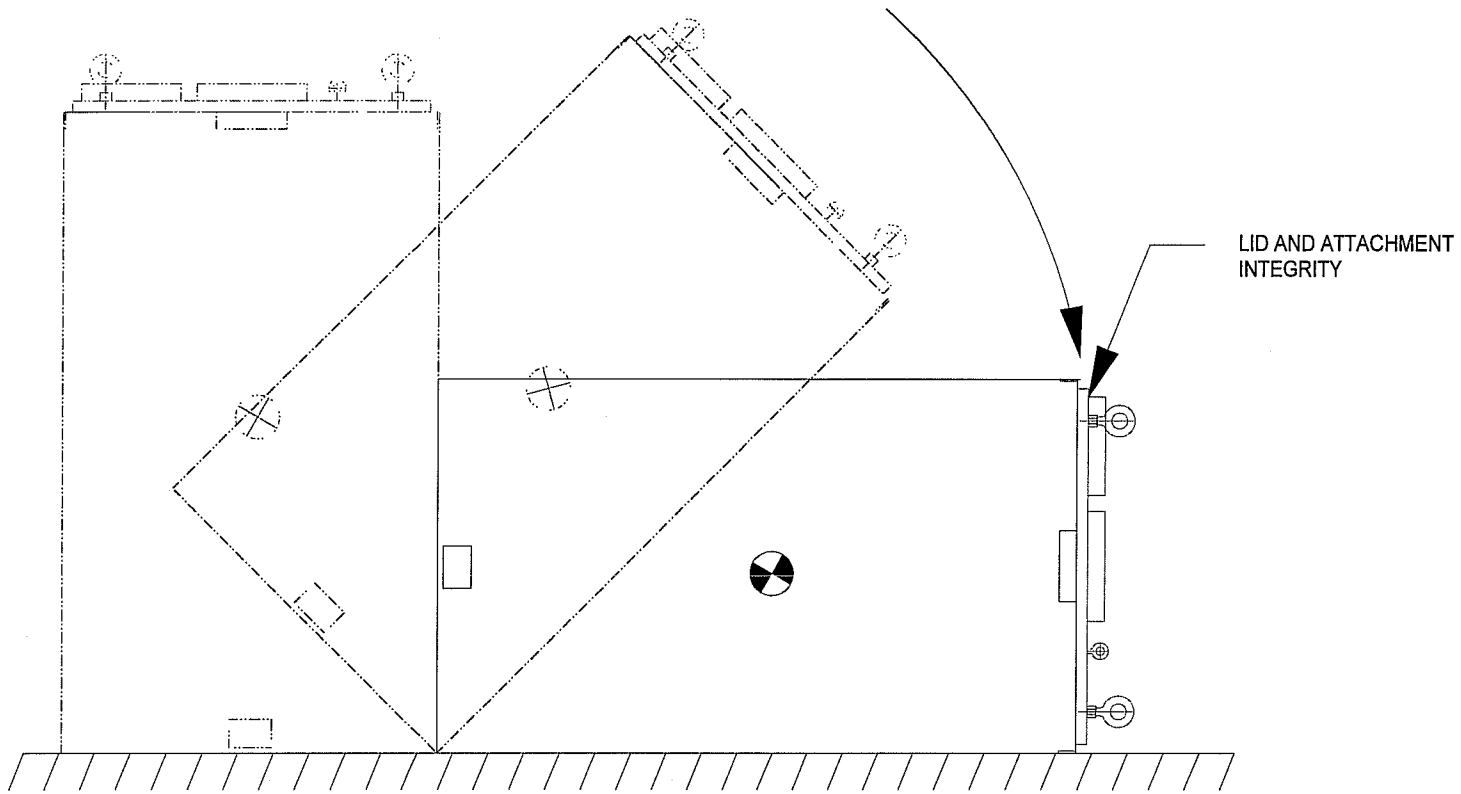




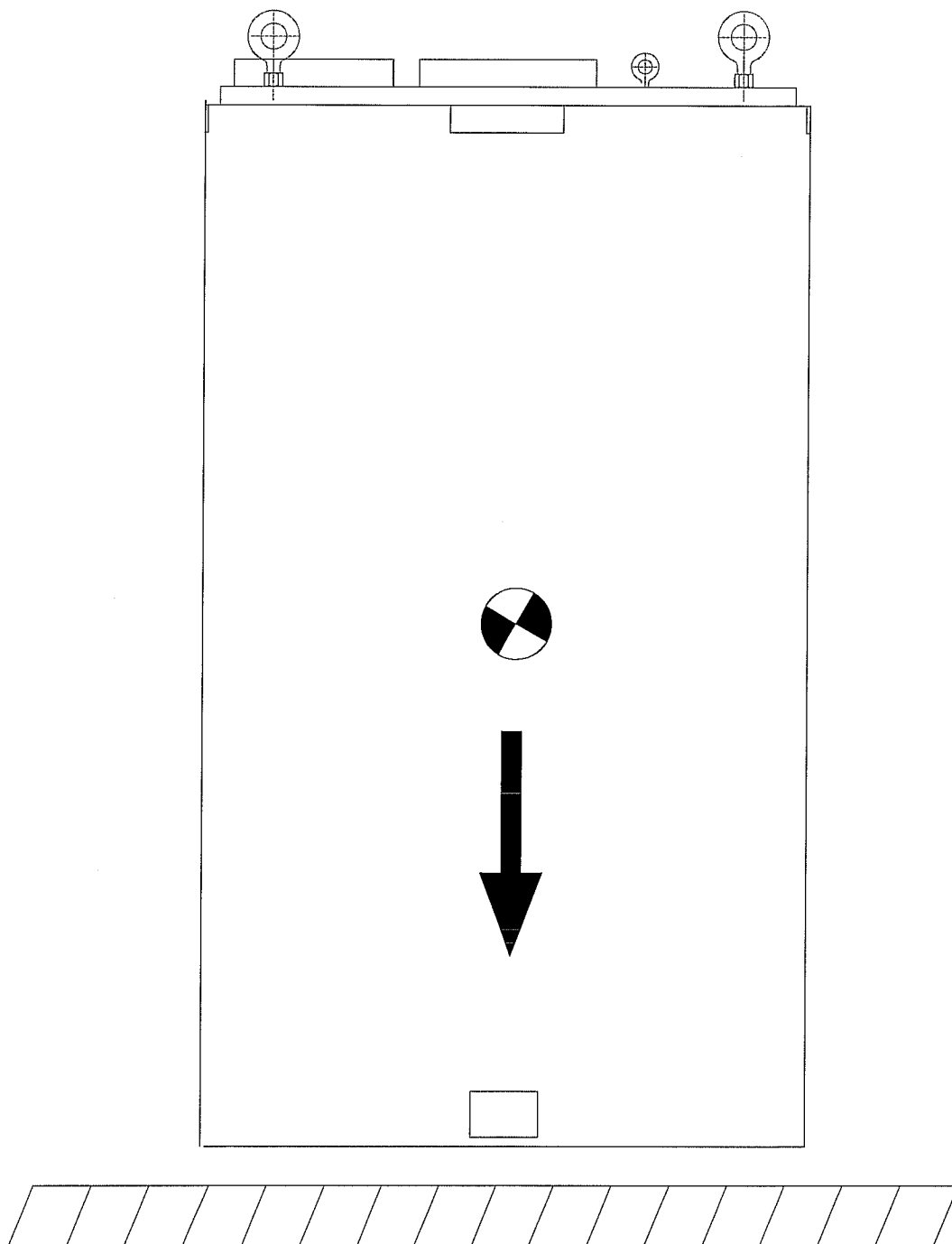
**Figure 3.4.17; Top and Bottom Lifting of the Loaded HI-STORM 100**



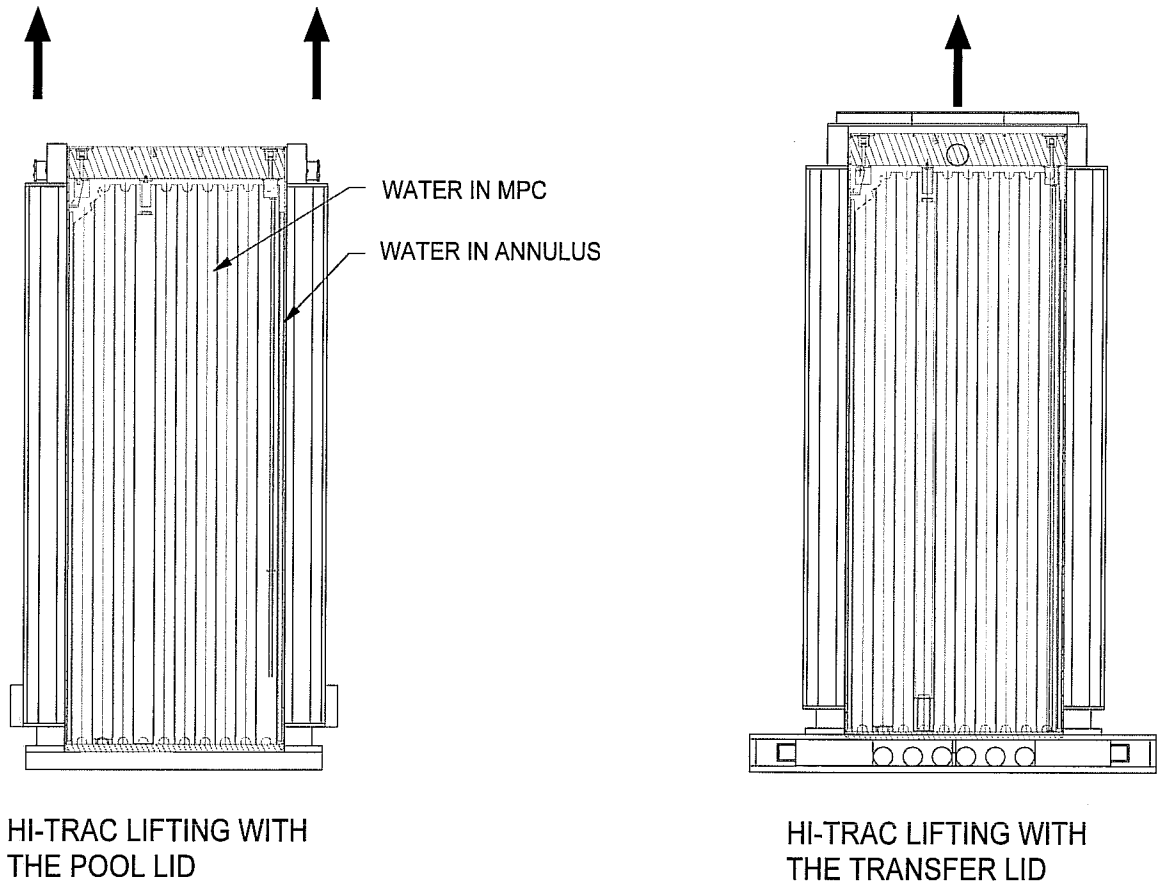
**Figure 3.4.18; HI-TRAC Upending in the Upending Frame**



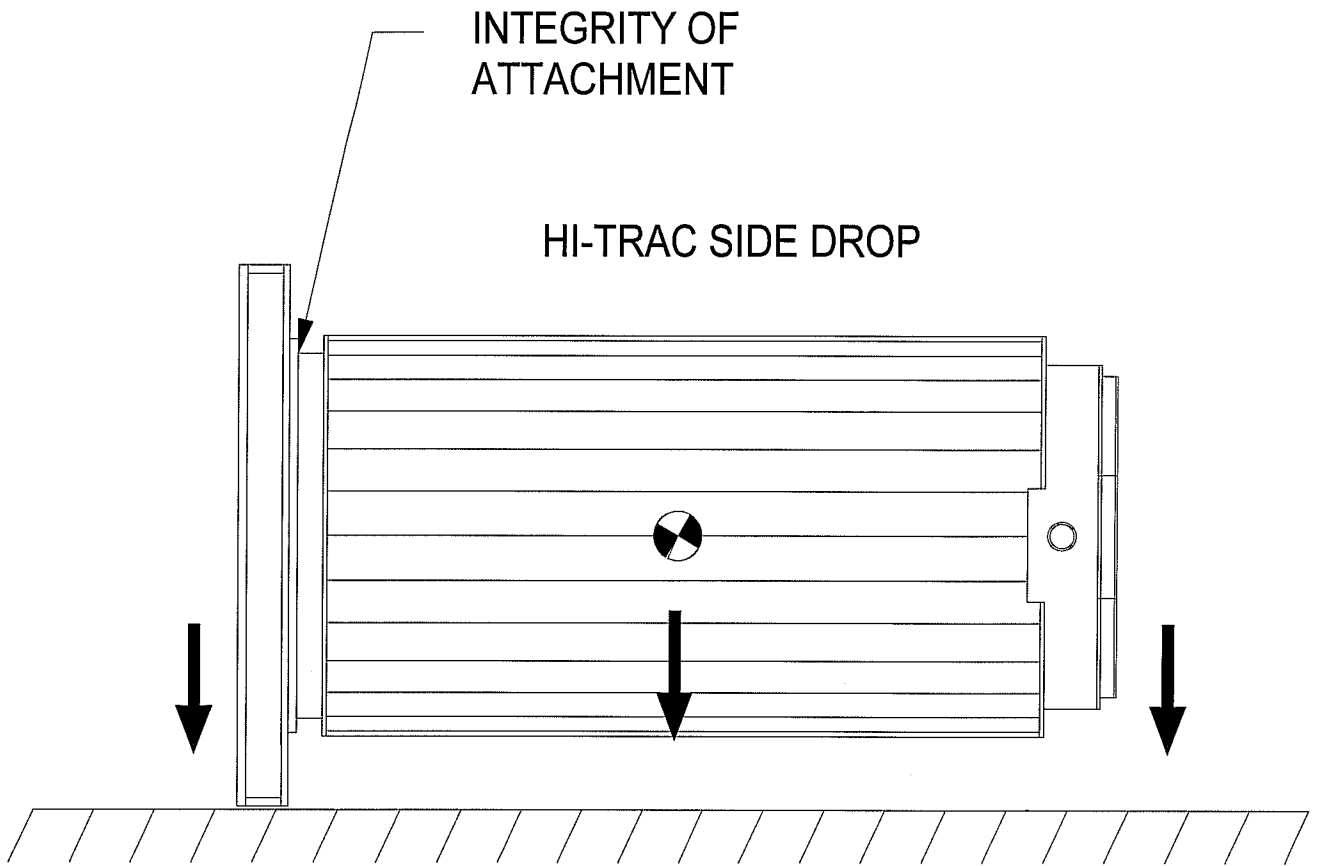
**Figure 3.4.19; HI-STORM 100 Tip-Over Event**



**Figure 3.4.20; HI-STORM 100 End Drop Event**



**Figure 3.4.21; HI-TRAC Lifting with the Pool and Transfer Lids**



**Figure 3.4.22; HI-TRAC Side Drop Event**

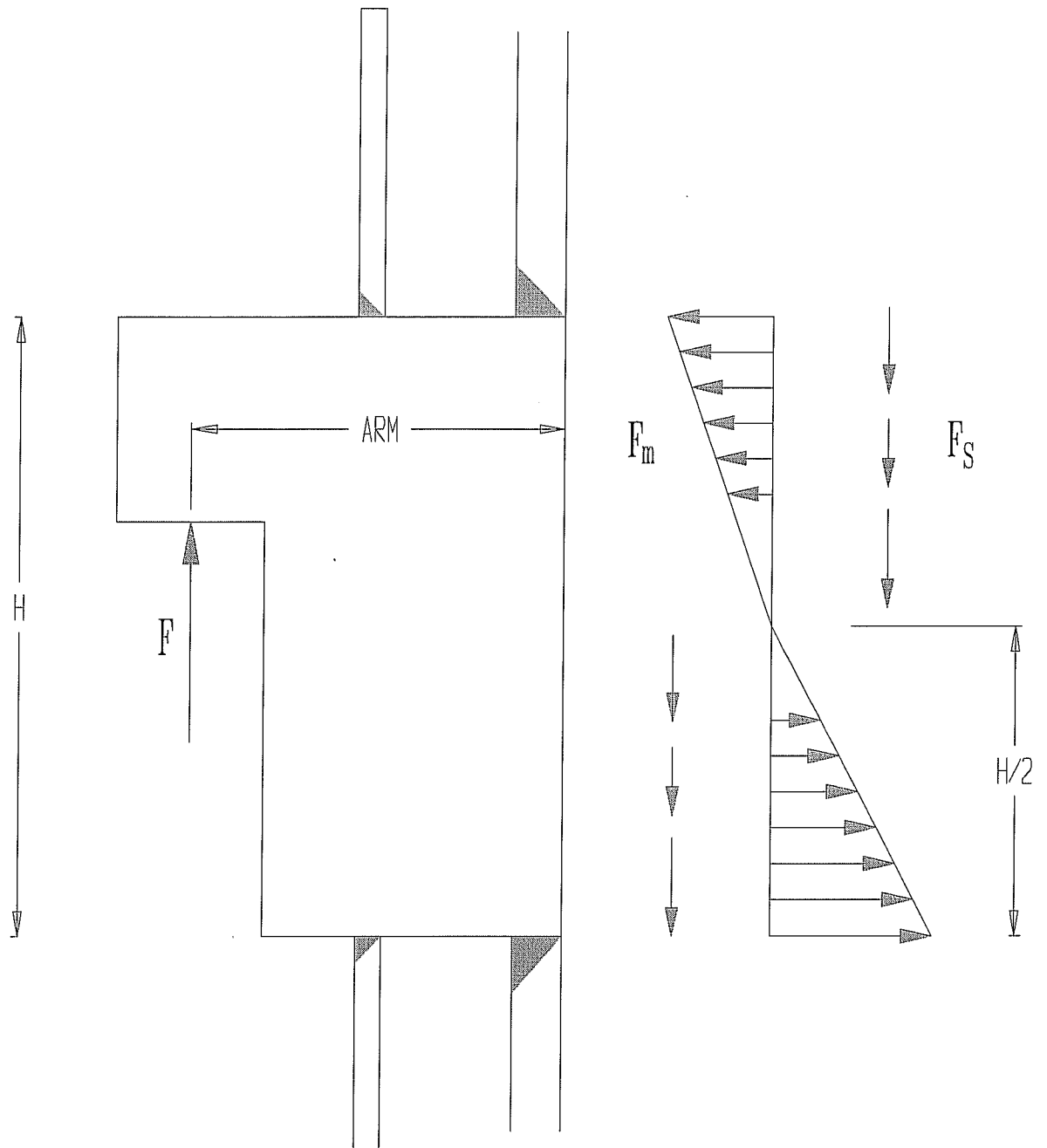


FIGURE 3.4.23 FORCES AND MOMENTS ON  
125 TON ROTATION TRUNNION WELD

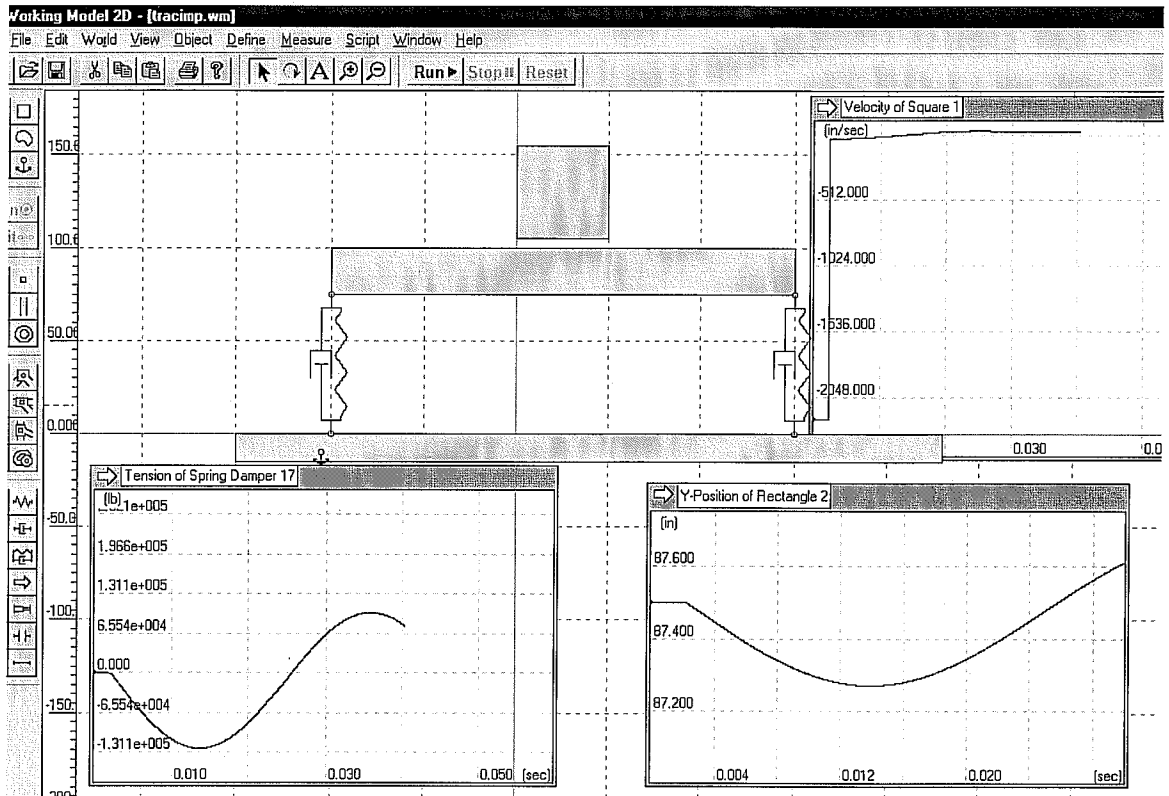


FIGURE 3.4.24 WORKING MODEL SOLUTION FOR IMPACT FORCE ON HI-TRAC 100 TRANSFER CASK OUTER SHELL



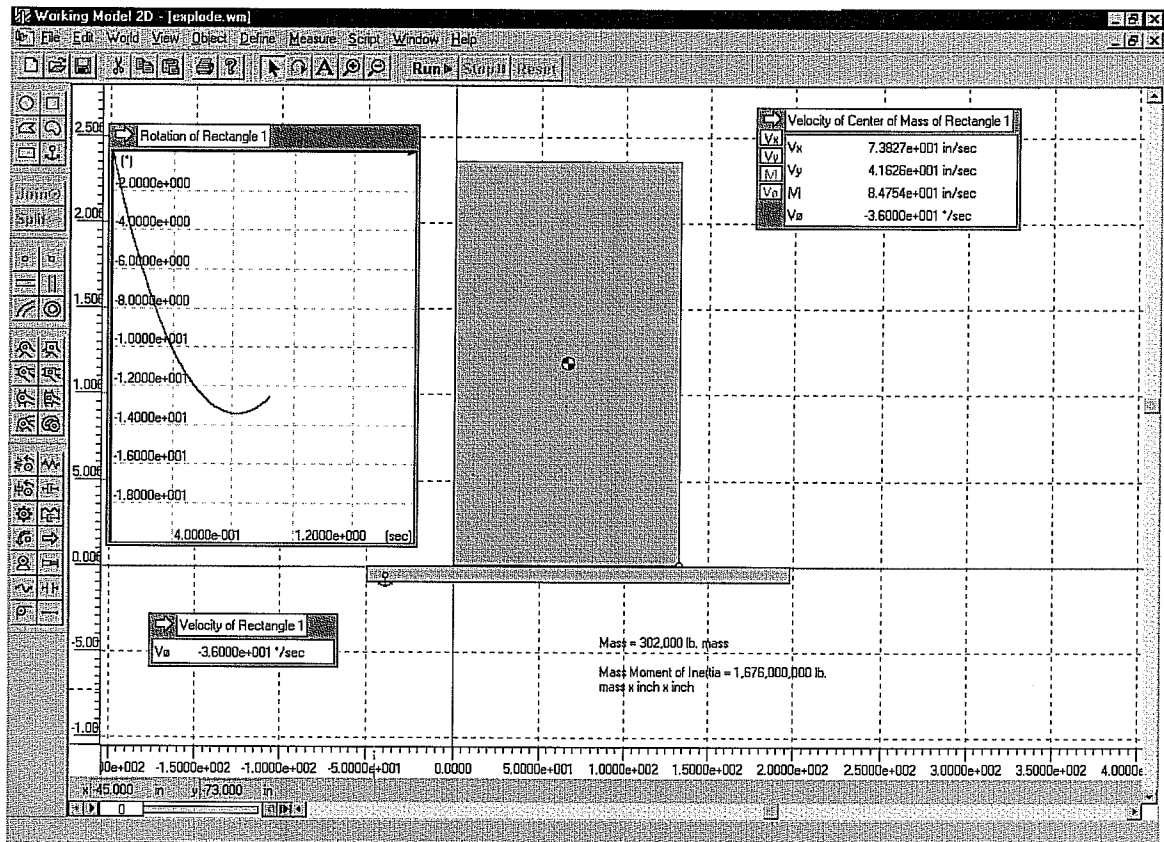


FIGURE 3.4.25: HI-STORM 100 OVERTURNING SCENARIO - INITIAL ANGULAR VELOCITY = 0.628 RADIANS/SECOND ASSUMED CAUSED BY A PRESSURE PULSE

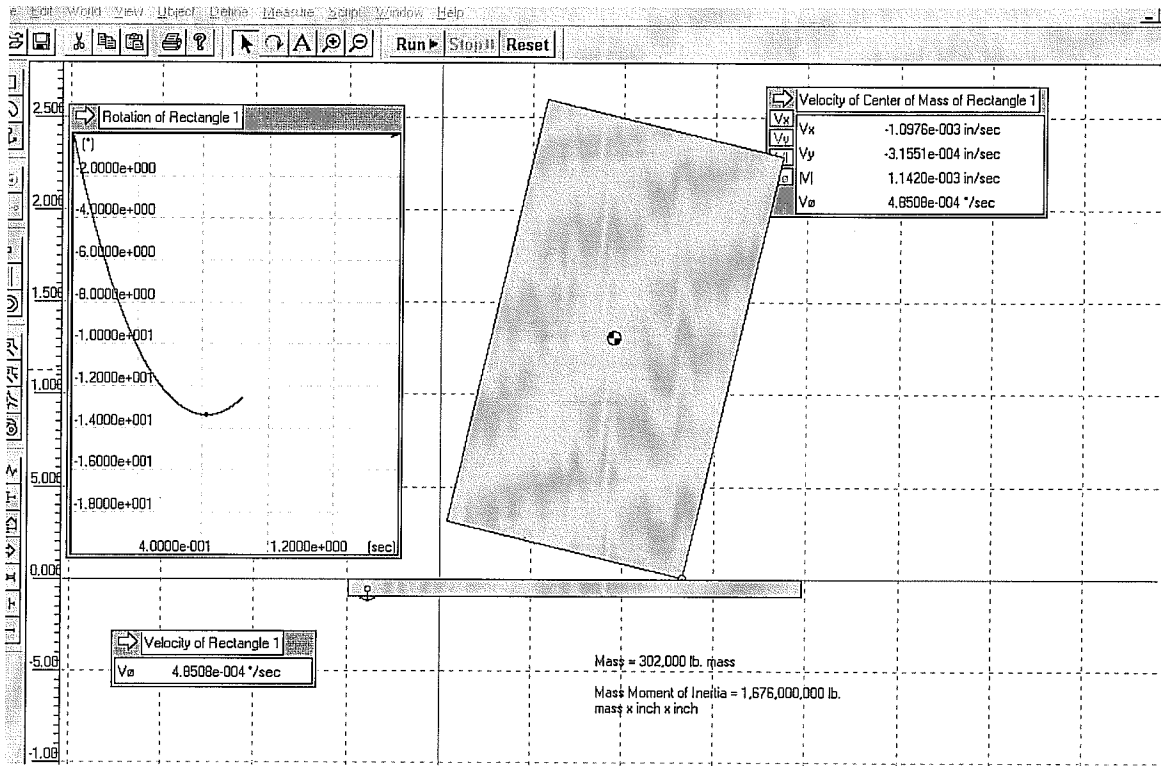
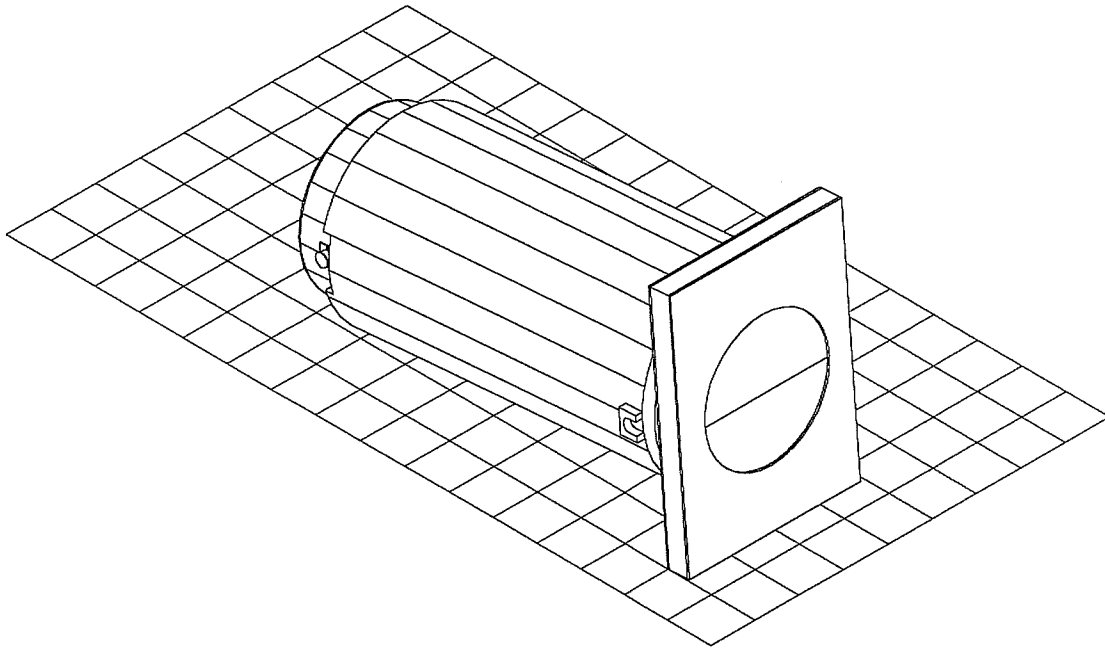
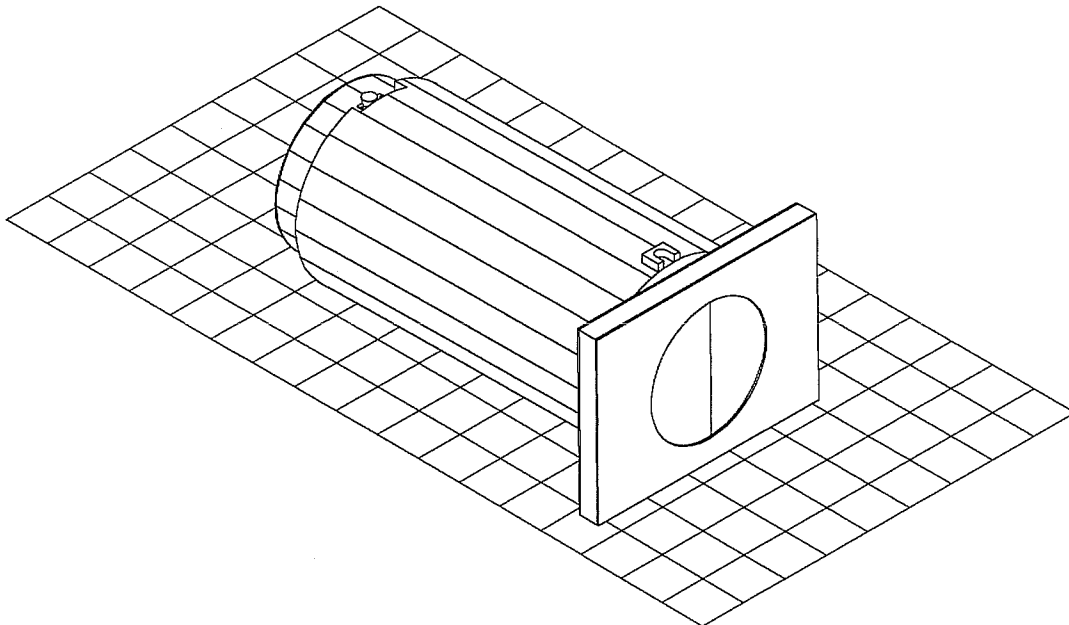


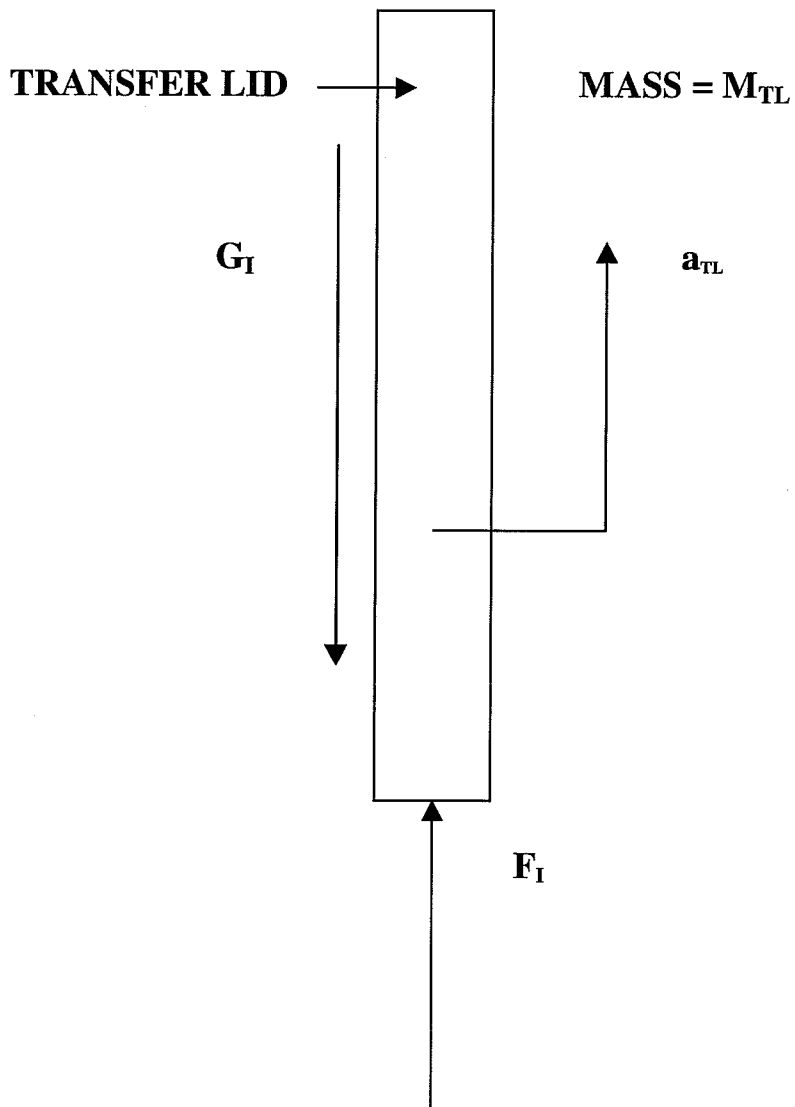
FIGURE 3.4.26: HI-STORM 100 OVERTURNING SCENARIO - INITIAL ANGULAR VELOCITY = 0.628 RADIANS/SECOND MAXIMUM ANGULAR EXCURSION



**FIGURE 3.4.27; HI-TRAC TRANSFER CASK IN SHORT-SIDE IMPACT  
(CASK RESTS AT A POSITION OF  $-5^{\circ}$  FROM HORIZONTAL)**



**FIGURE 3.4.28; HI-TRAC TRANSFER CASK IN LONG-SIDE IMPACT  
(CASK RESTS AT A POSITION OF  $-1^{\circ}$  FROM HORIZONTAL)**



**FIGURE 3.4.29; FREE-BODY OF TRANSFER LID DURING PRIMARY IMPACT WITH TARGET**

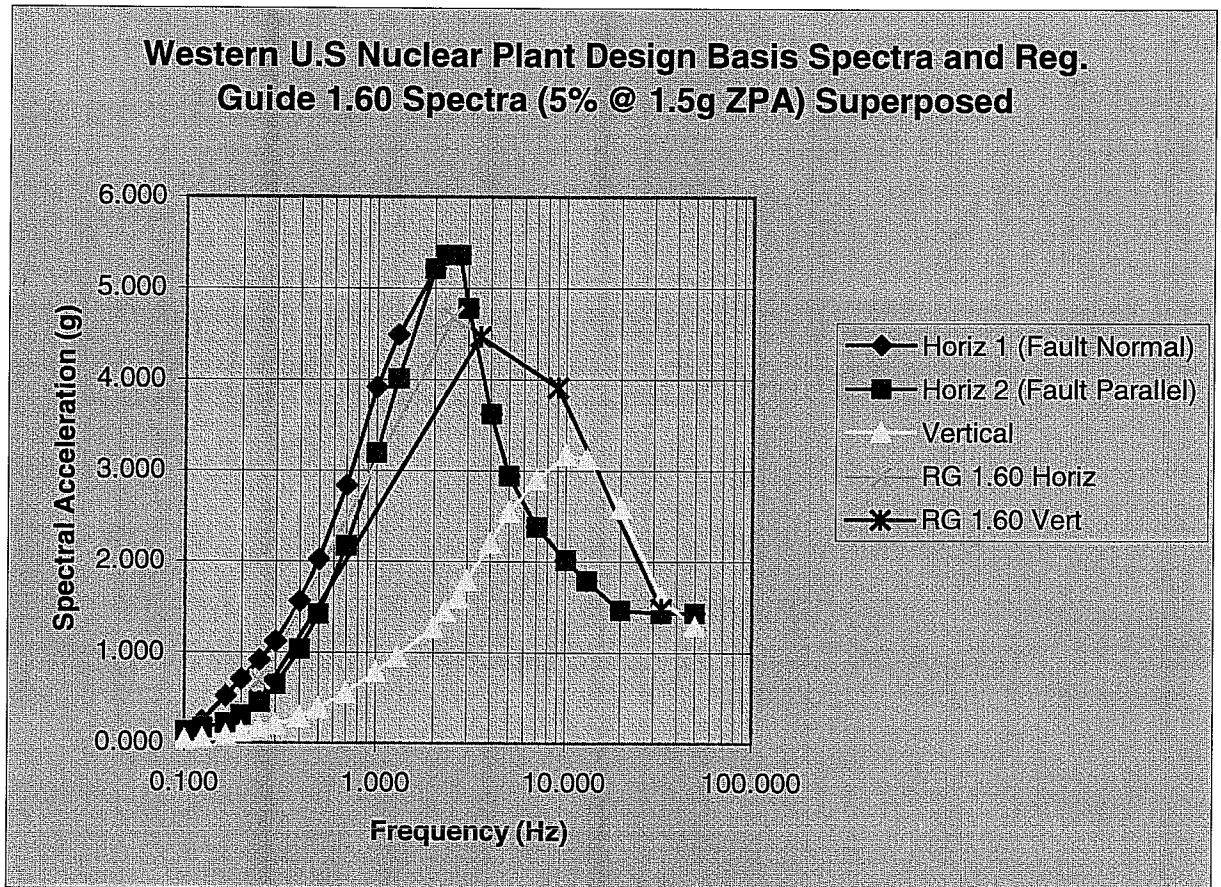


FIGURE 3.4.30 SEISMIC SPECTRA SETS USED FOR TIME HISTORY ANALYSIS OF HI-STORM 100A ON ISFSI PAD

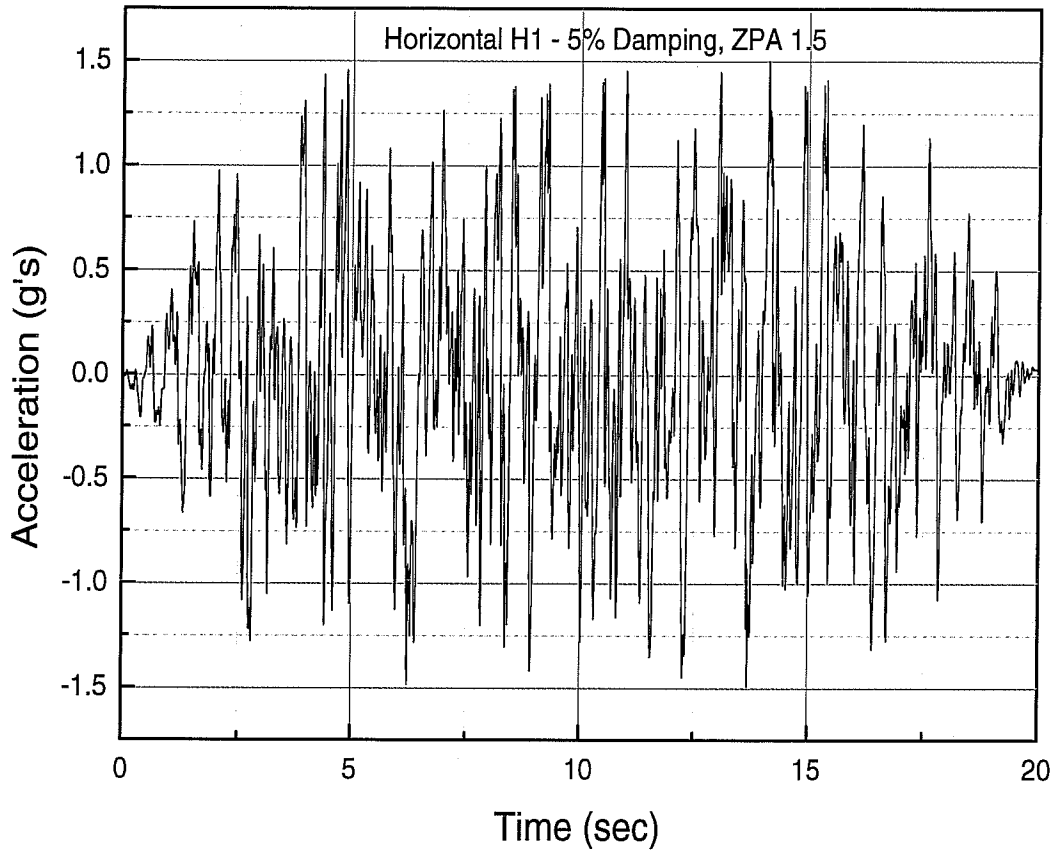


FIGURE 3.4.31 – RG 1.60 “H1”

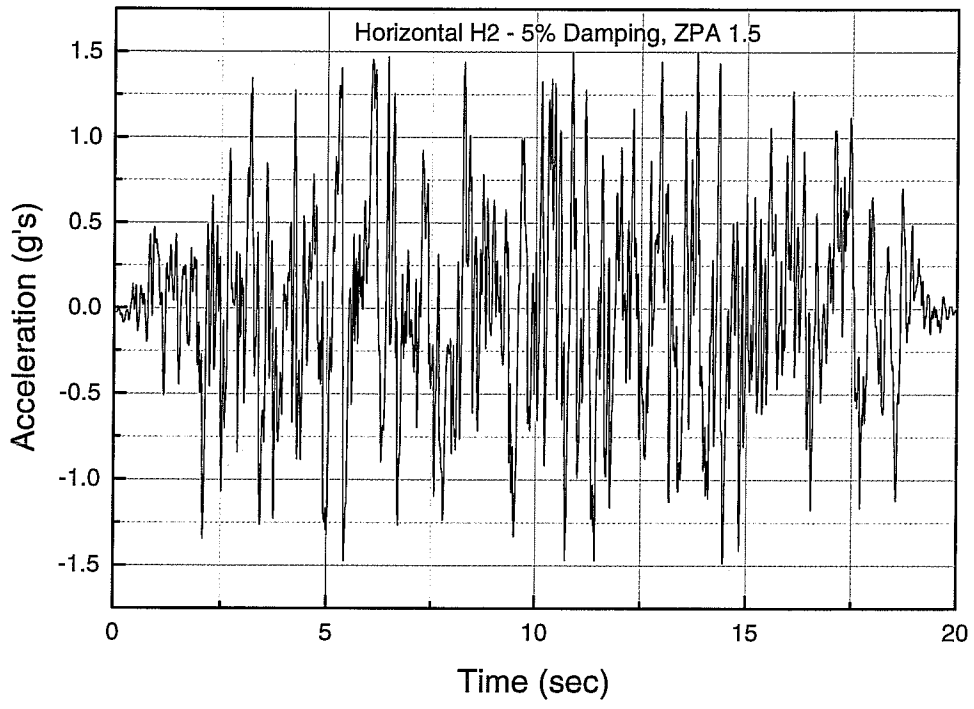


FIGURE 3.4.32 – RG 1.60 “H2”

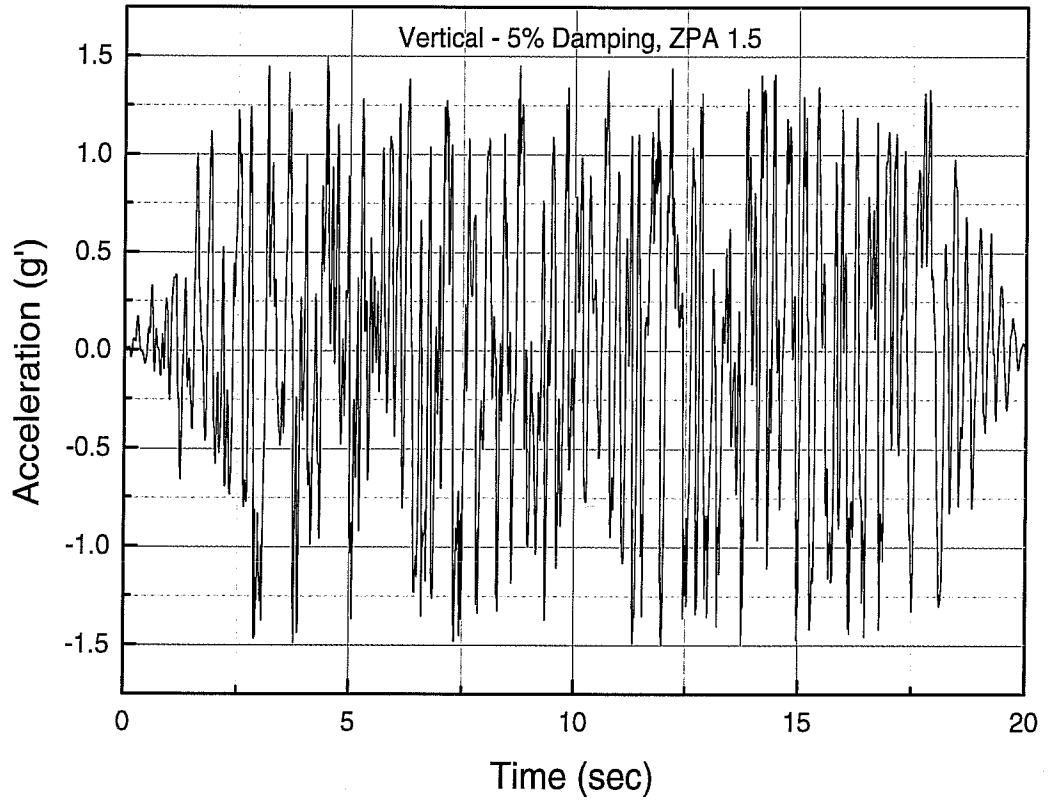


FIGURE 3.4.33 – RG 1.60 “VT”



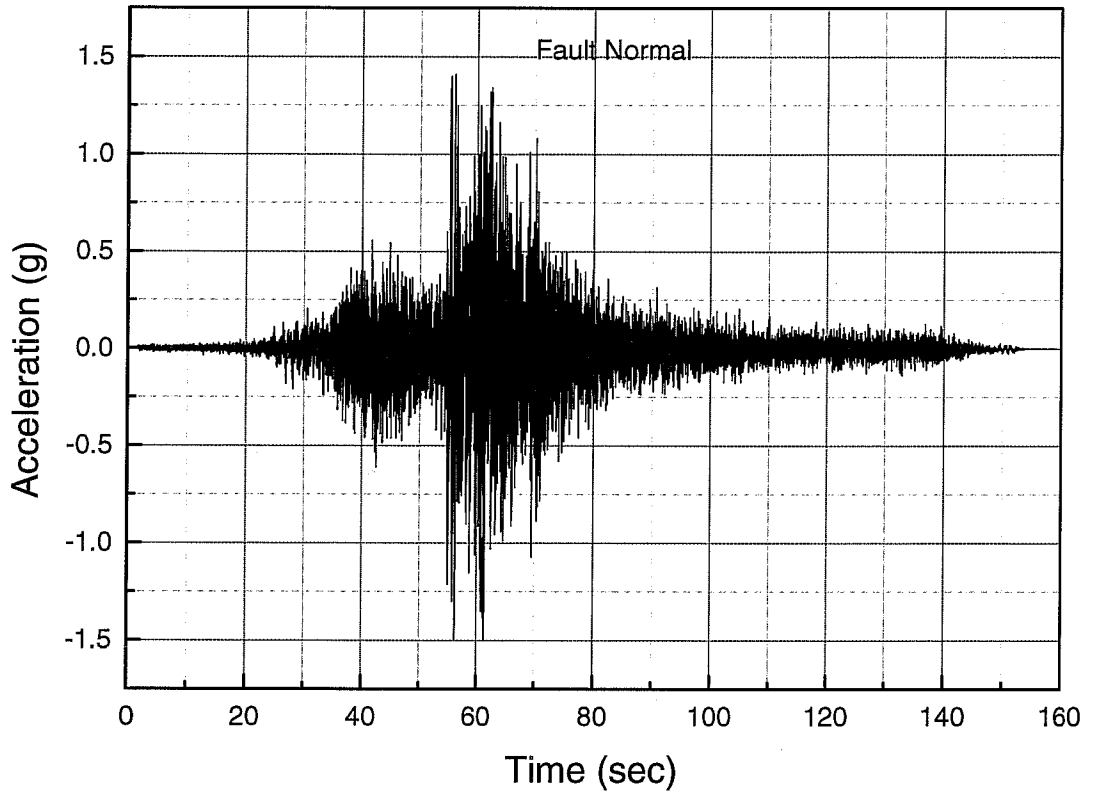


FIGURE 3.4.34 Horizontal Acceleration Time history "FN"

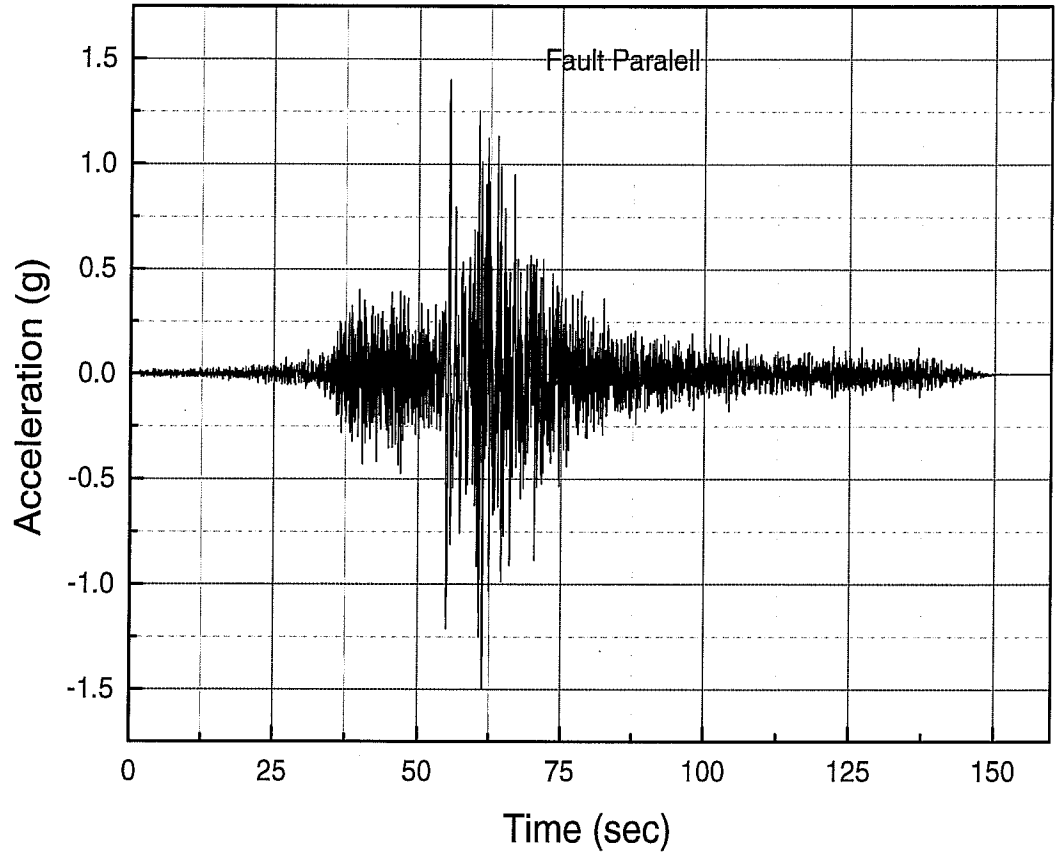


FIGURE 3.4.35 Horizontal Acceleration Time history "FP"

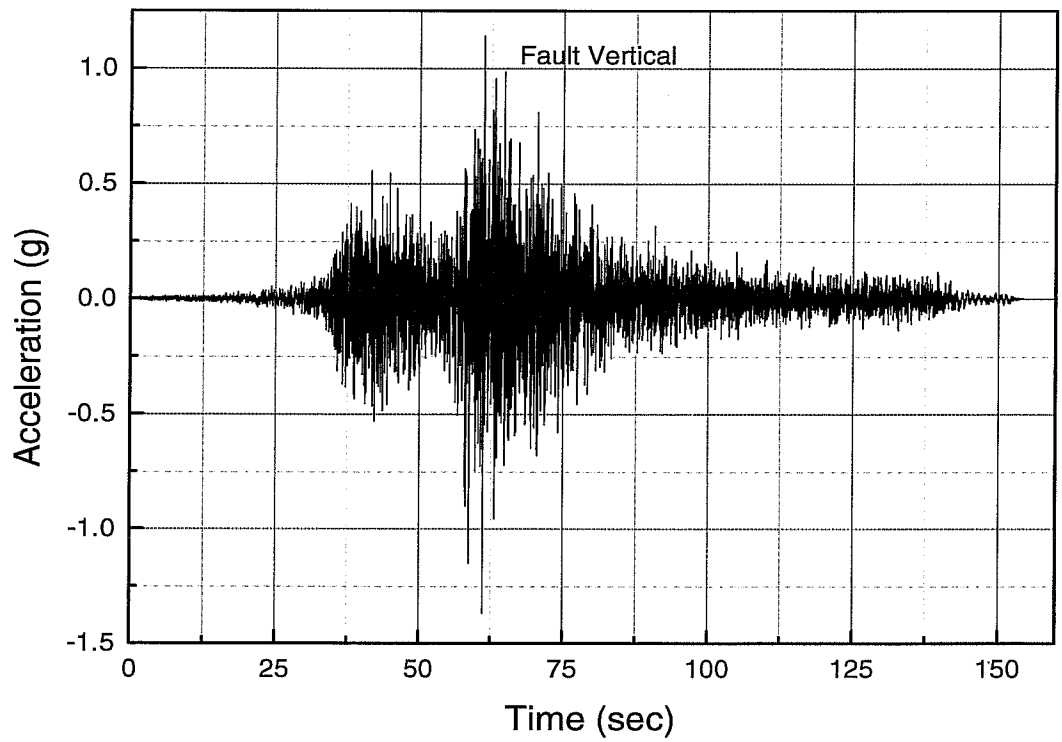


FIGURE 3.4.36 Vertical Acceleration Time history "FV"

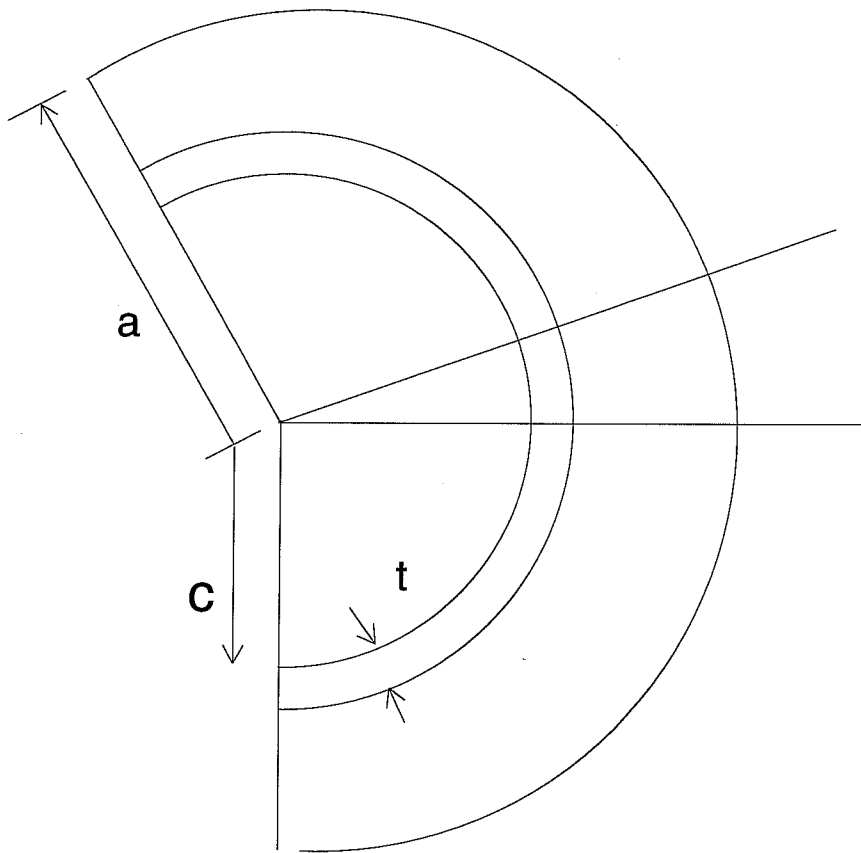


FIGURE 3.4.37 GEOMETRY FOR QUASI-STATIC ANALYSIS

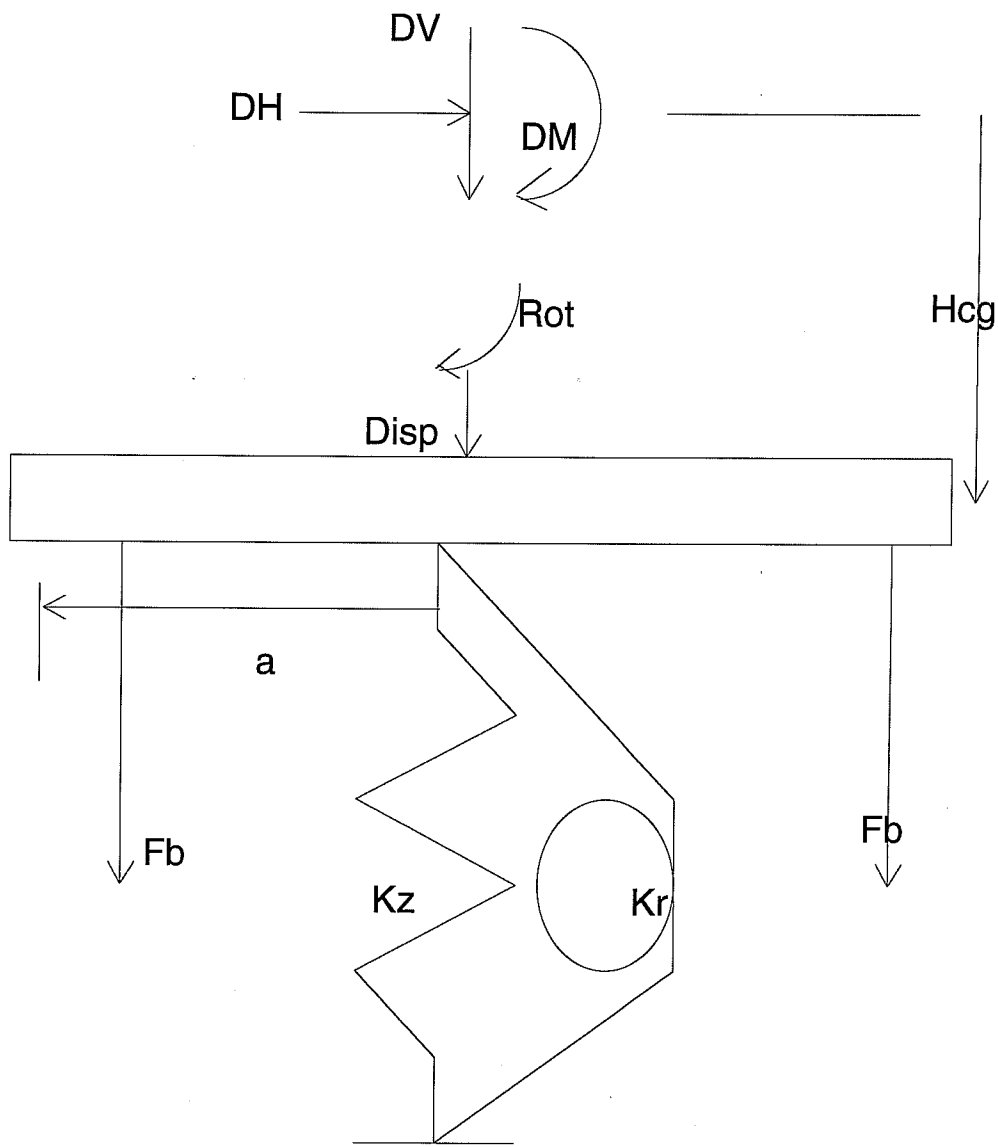


FIGURE 3.4.38 FREE BODY FOR QUASI-STATIC ANALYSIS

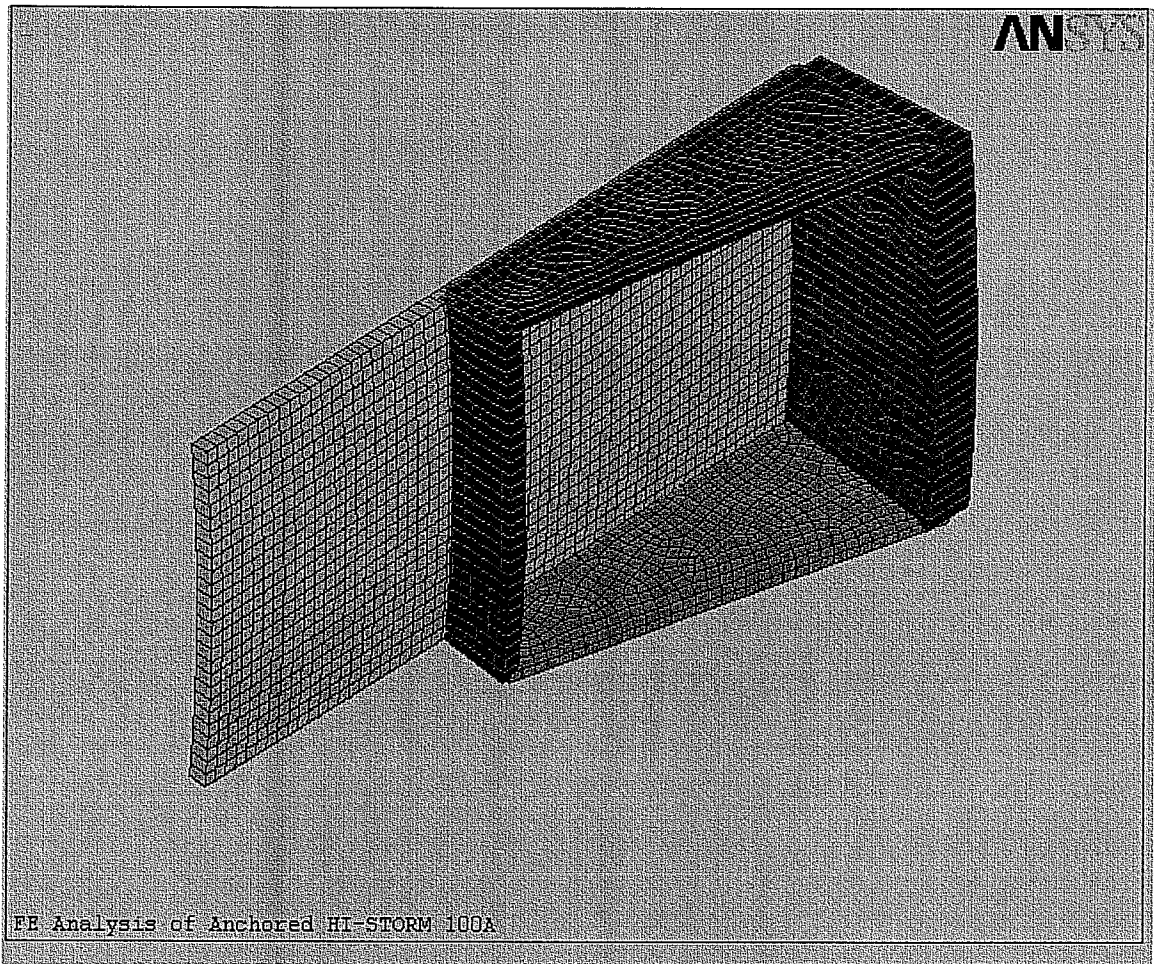


FIGURE 3.4.39 Sector Lug Finite Element Mesh

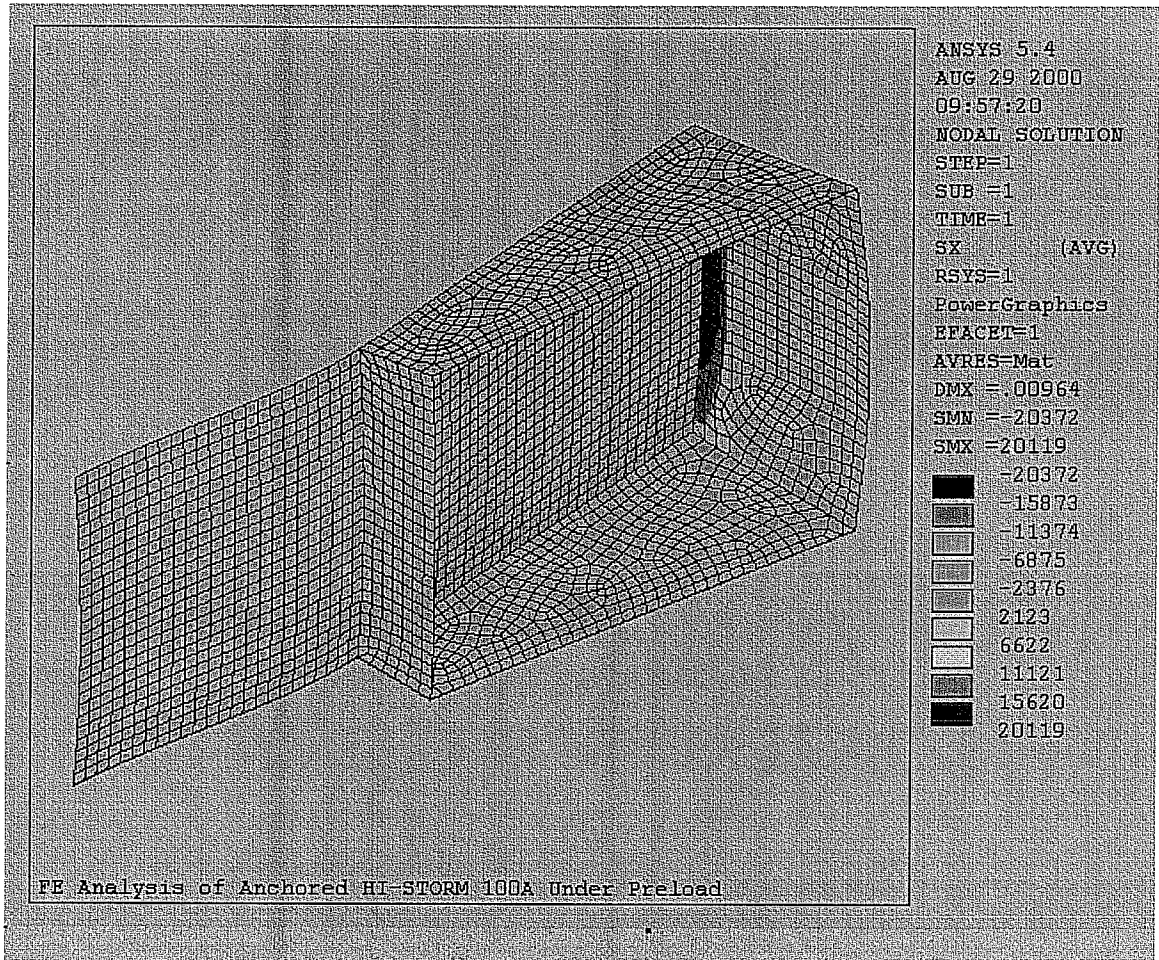


FIGURE 3.4.40 Sector Lug Stress – Case 1 Preload

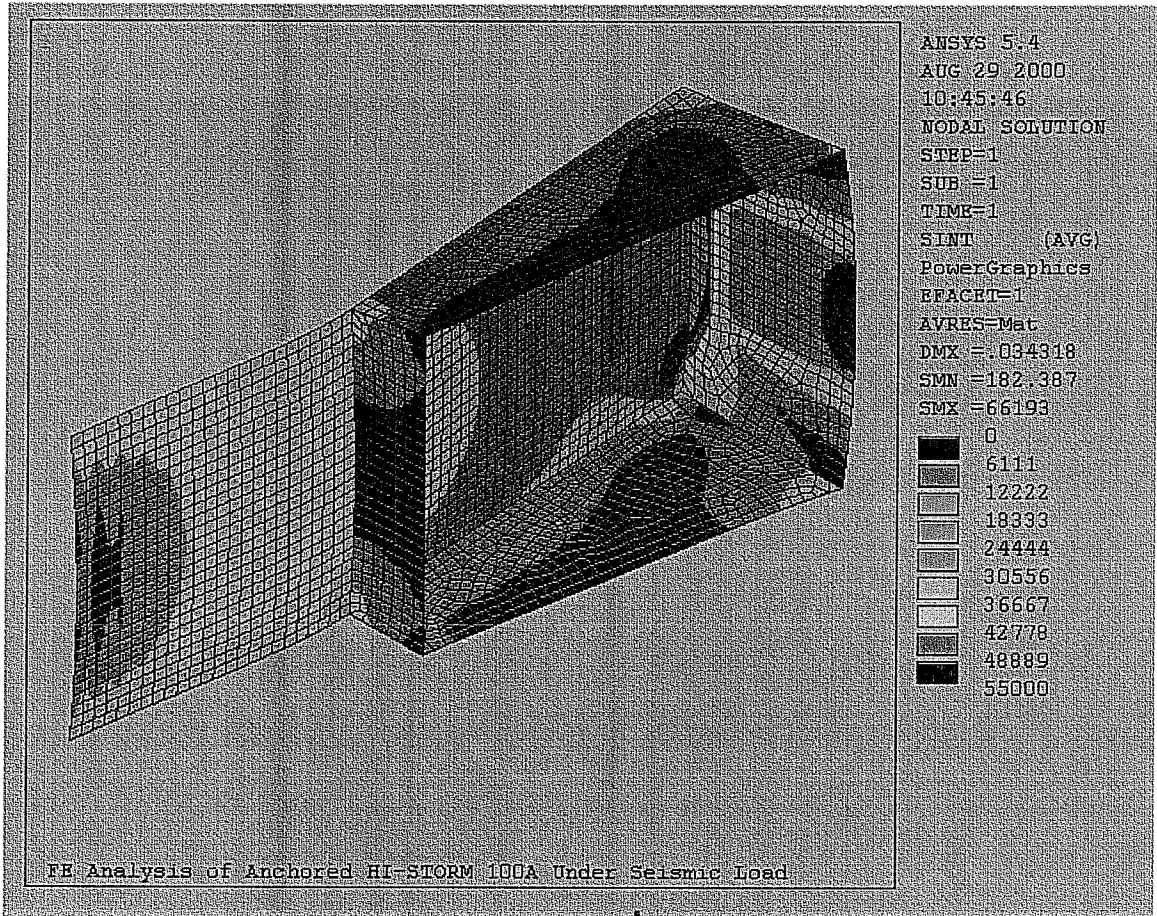


FIGURE 3.4.41 Sector Lug Stress Intensity – Case 2 Preload + Seismic



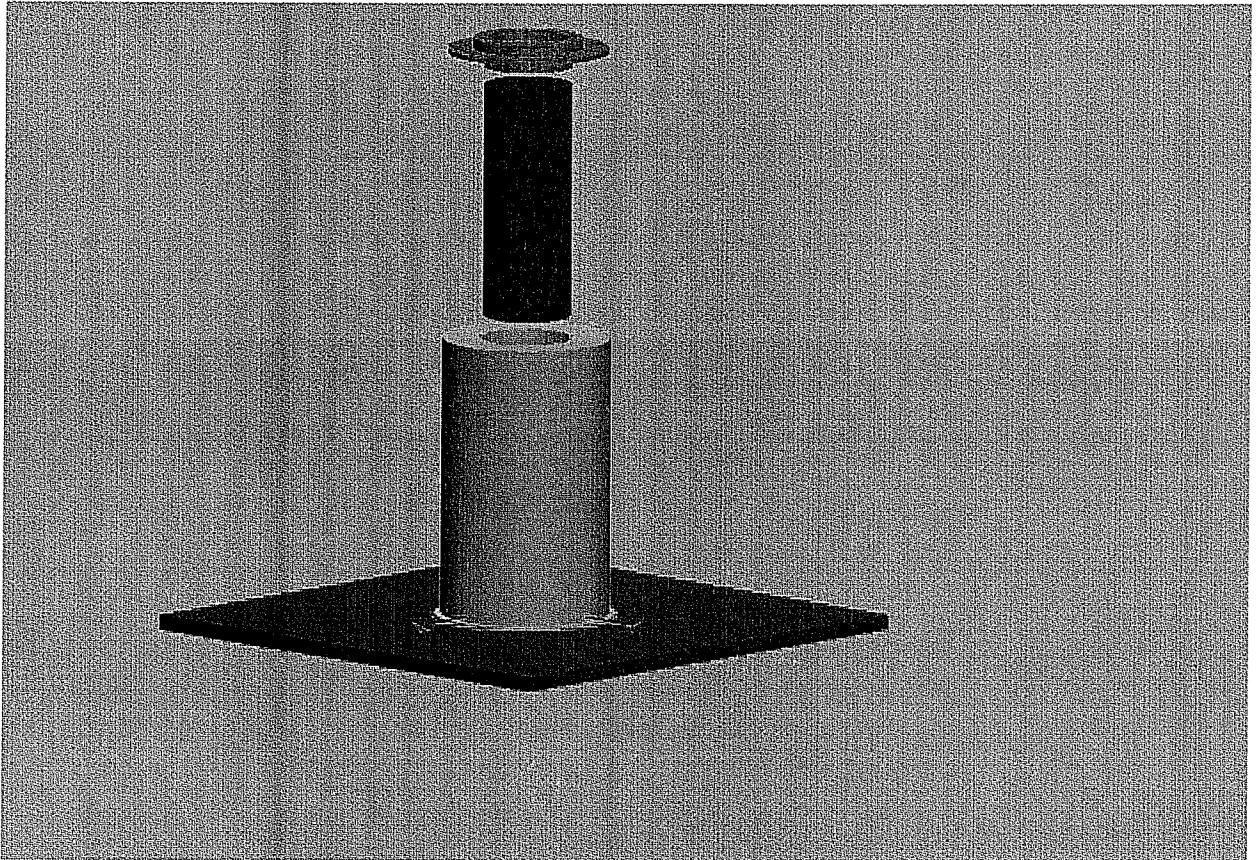
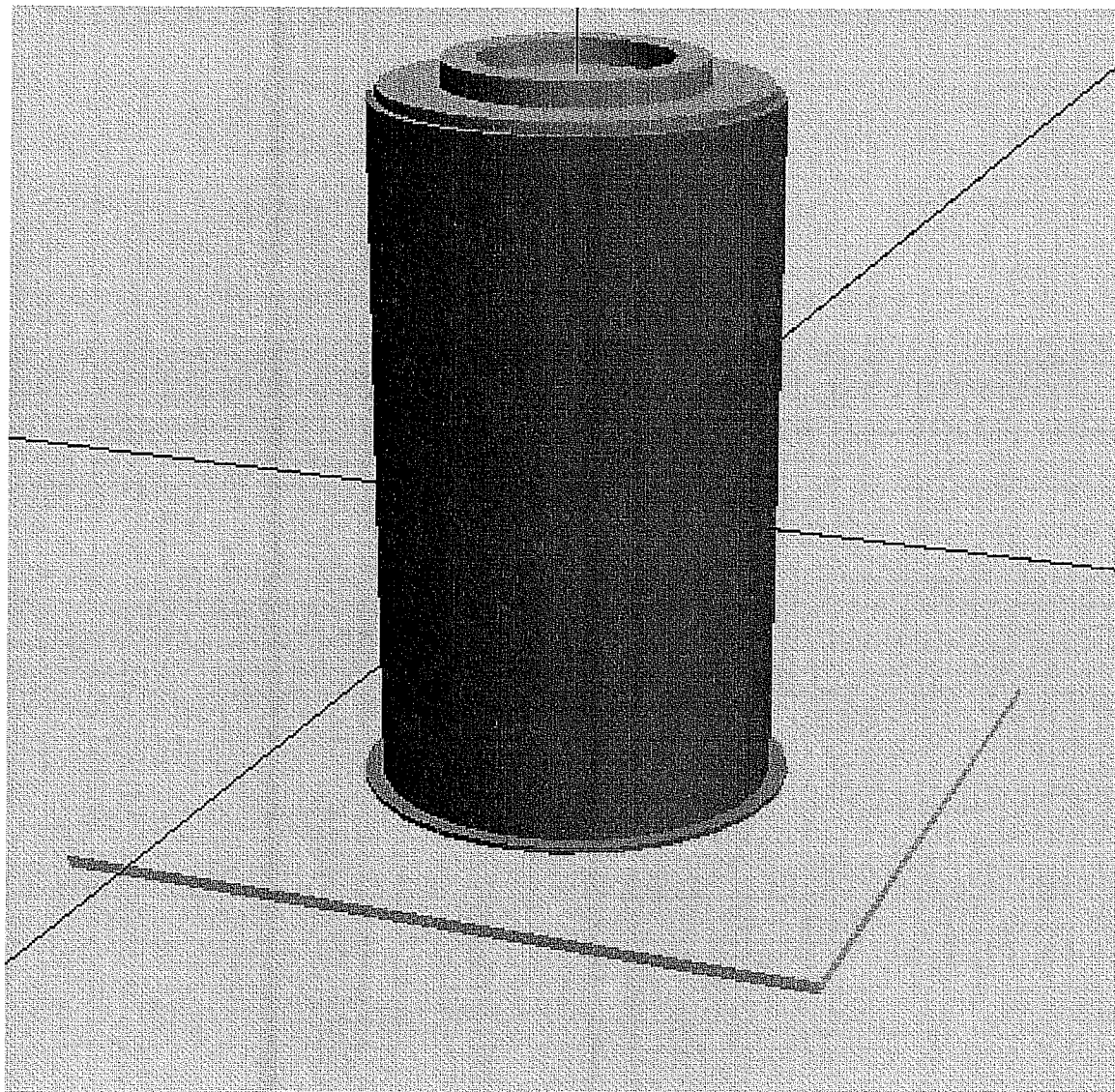


FIGURE 3.4.42: EXPLODED VIEW SHOWING GROUND PLANE, OVERPACK, MPC, AND OVERPACK TOP LID



**FIGURE 3.4.43: VIEW OF ASSEMBLED HI-STORM ON PAD-MPC INSIDE AND TOP LID ATTACHED (Note Extended Baseplate for Anchor Connections)**

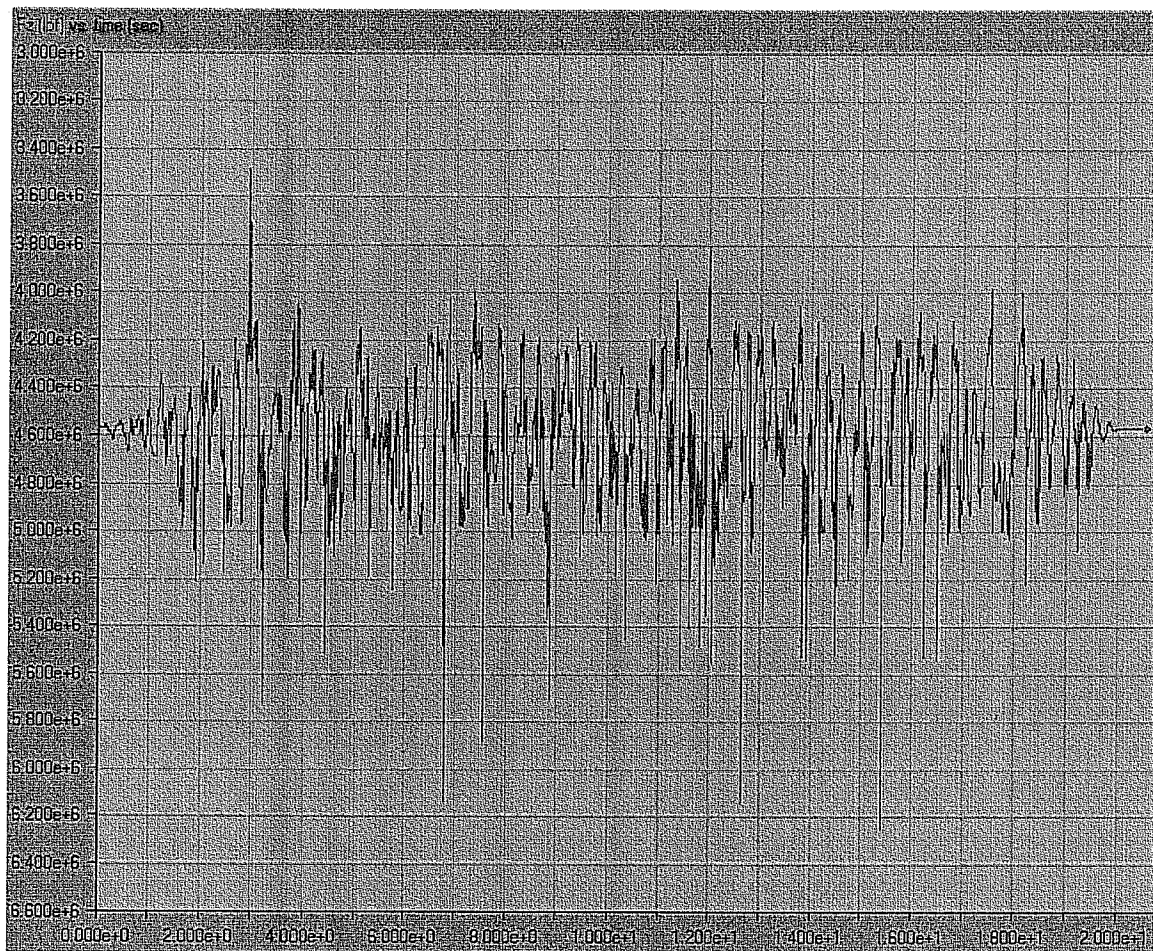


FIGURE 3.4.44 Variation of Foundation Resistance Force vs. Time for Reg. Guide 1.60 Seismic Input

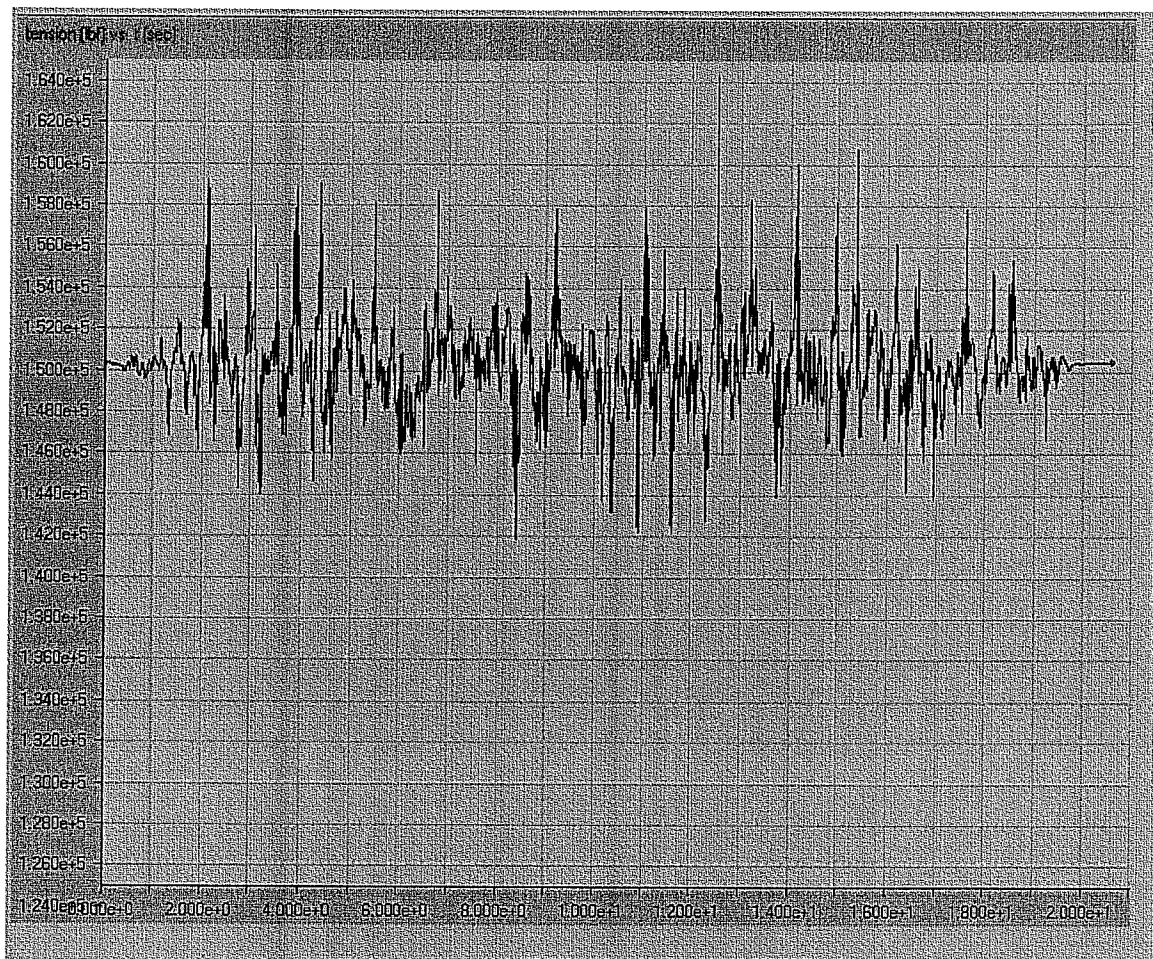


FIGURE 3.4.45 Variation of Representative Stud Tensile Force vs. Time for Reg. Guide 1.60 Seismic Input

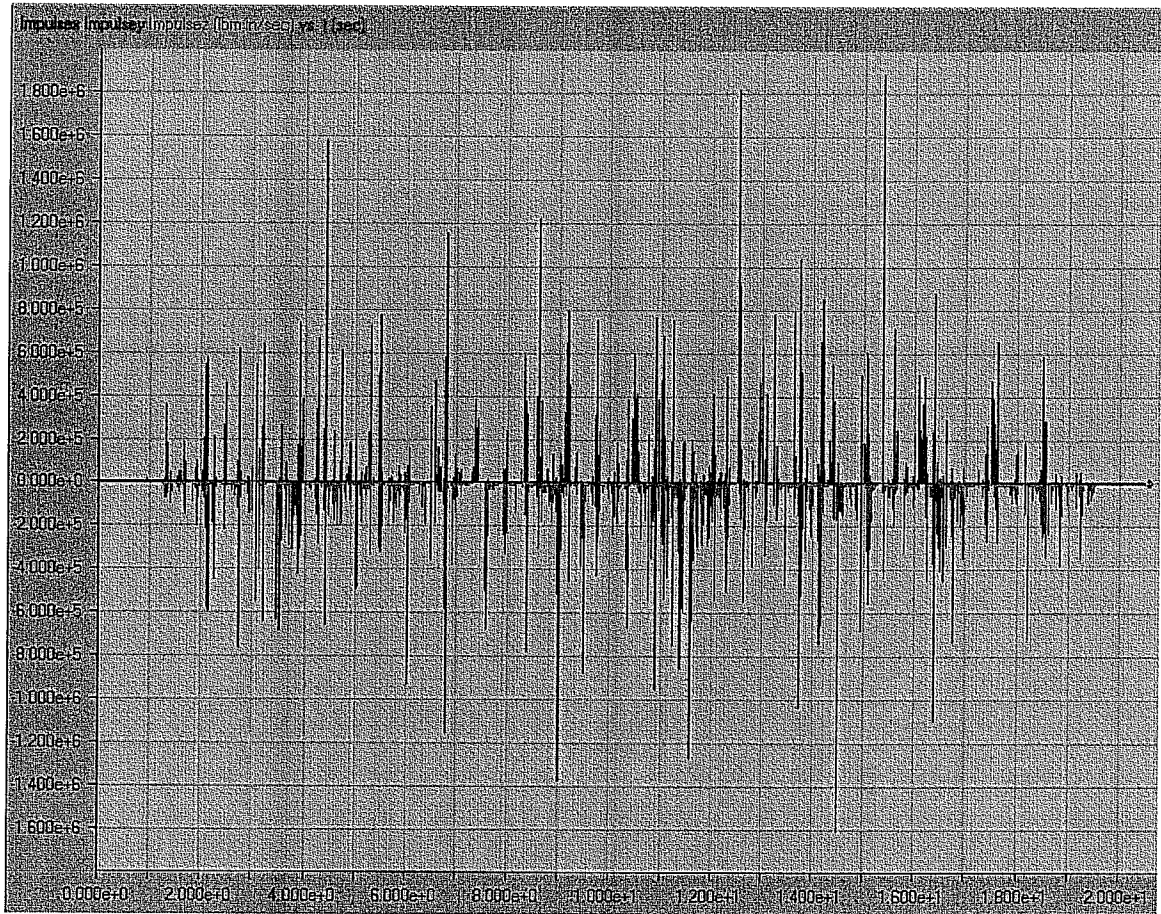


FIGURE 3.4.46 MPC/Hi-STORM 100A Impulse vs. Time – Reg. Guide 1.60 Event

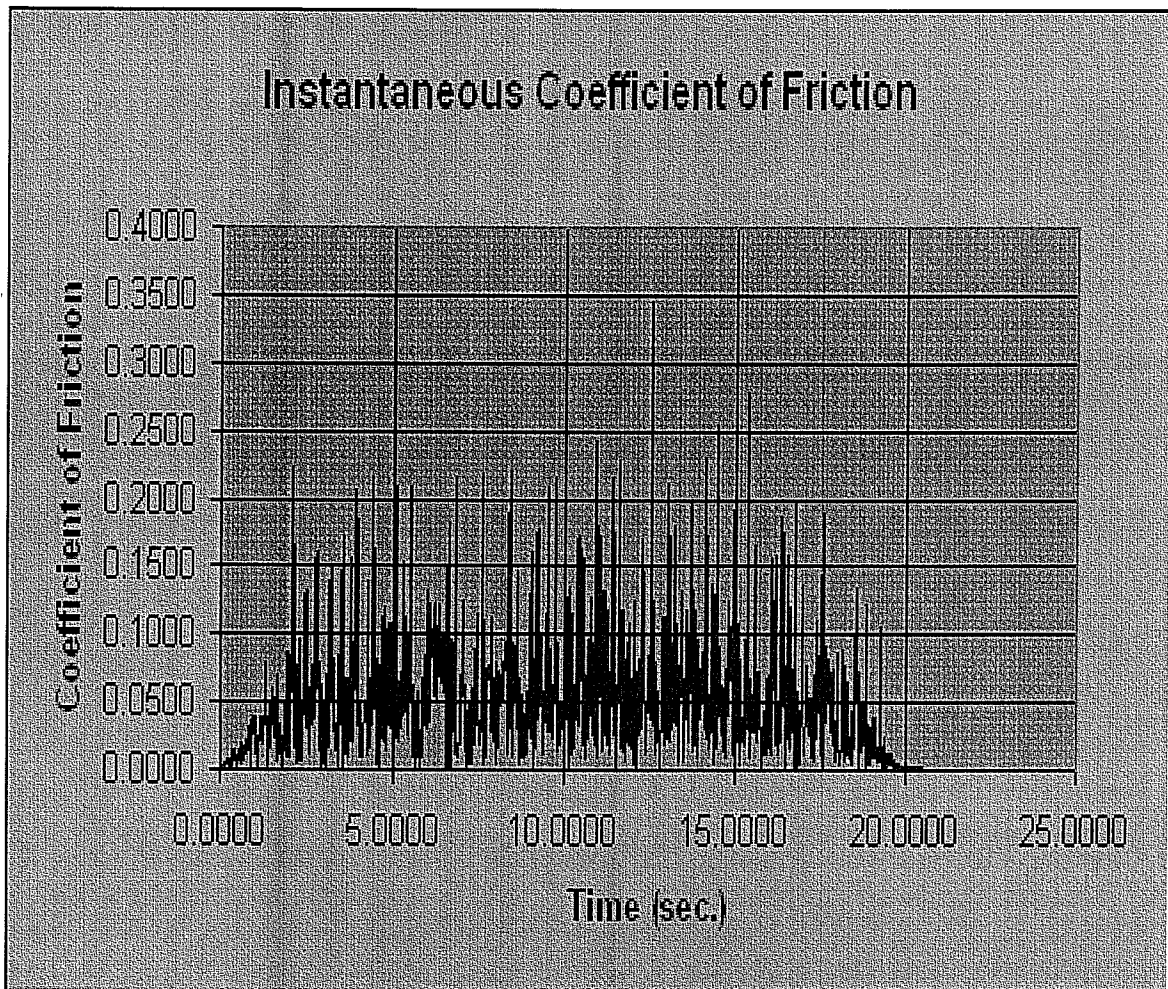
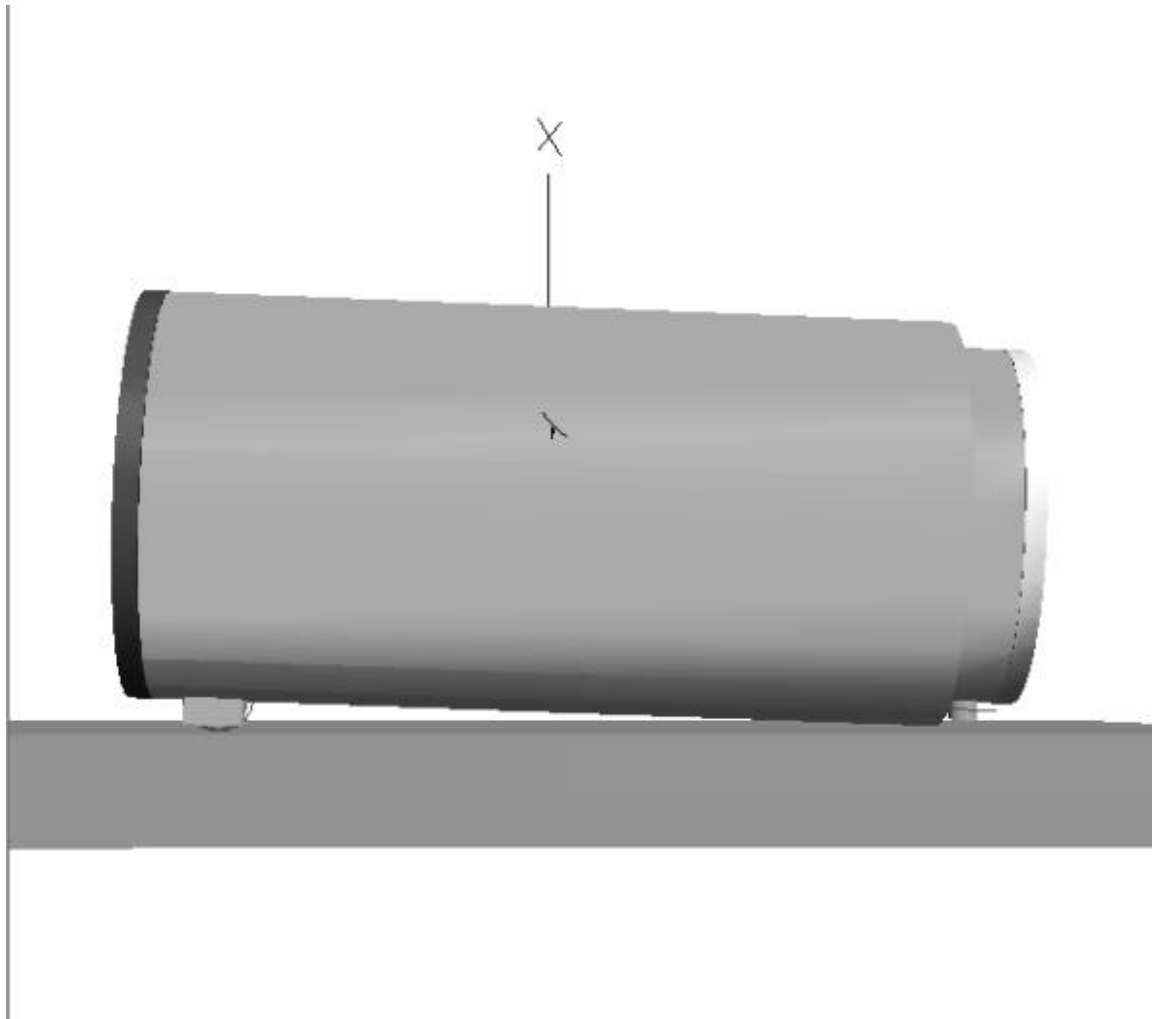
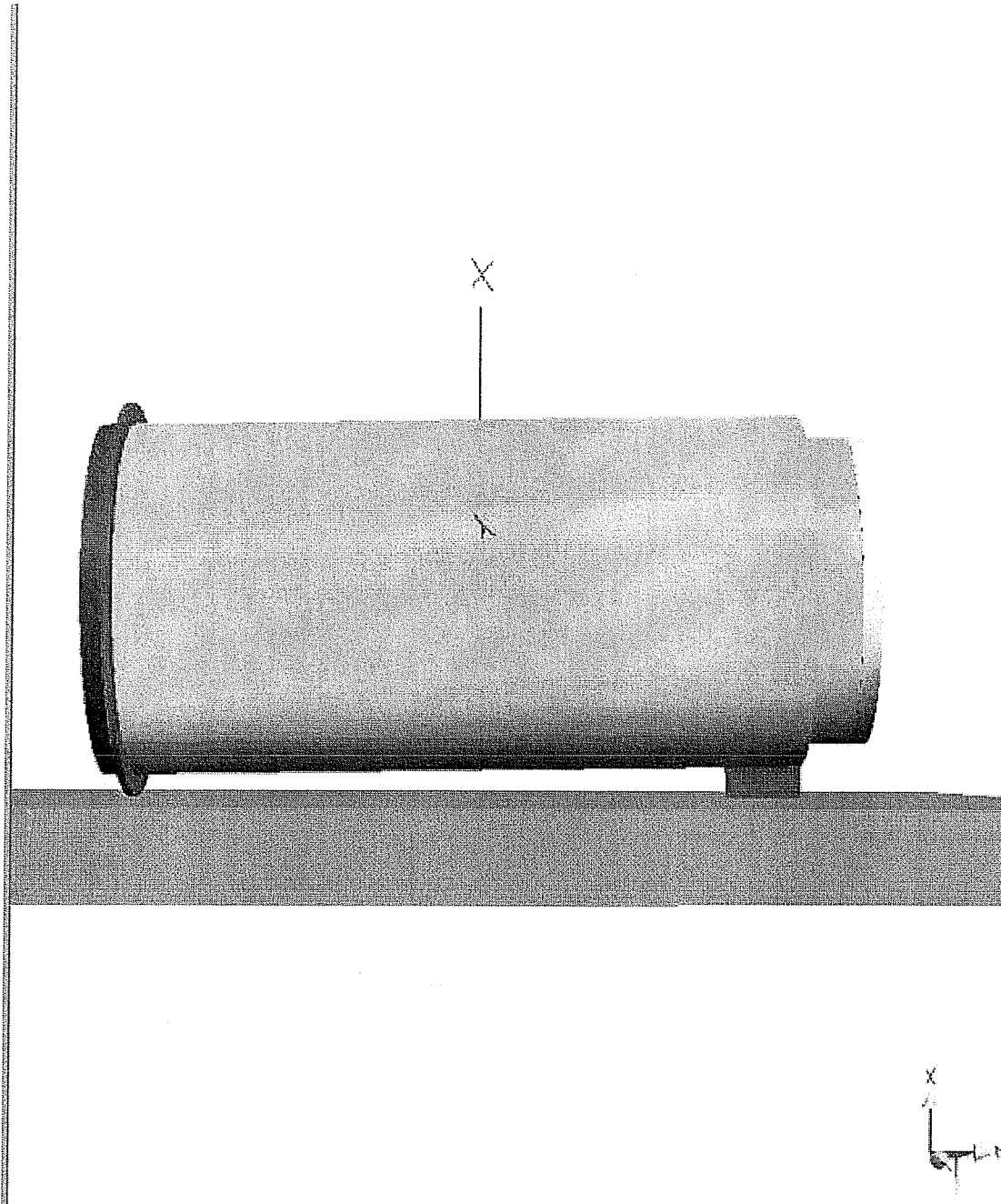


FIGURE 3.4.47 Instantaneous Calculated Coefficient of Friction – Reg. Guide 1.60 Event



**FIGURE 3.4.48; HI-TRAC 125 BENCHMARK SIMULATION OF DROP SCENARIO A**



**FIGURE 3.4.49; SIMULATION OF HI-TRAC 125D 42”  
HORIZONTAL DROP WITH PRIMARY IMPACT  
AT TOP END RADIAL SUPPORT TAB**



### 3.5 FUEL RODS

The cladding of the fuel rods is the initial confinement boundary in the HI-STORM 100 System. Analyses have been performed in Chapter 3 to ensure that the maximum temperature of the fuel cladding is below the Pacific Northwest Laboratory's threshold values for various cooling times. These temperature limits ensure that the fuel cladding will not degrade in an inert helium environment. Additional details on the fuel rod cladding temperature analyses for the spent fuel to be loaded into the HI-STORM 100 System are provided in Chapter 3.

The dimensions of the storage cell openings in the MPC are equal to or greater than those used in spent fuel racks supplied by Holtec International. Thousands of fuel assemblies have been shuffled in and out of these cells over the years without a single instance of cladding failure. The vast body of physical evidence from prior spent fuel handling operations provides confirmation that the fuel handling and loading operations with the HI-STORM 100 MPC will not endanger or compromise the integrity of the cladding or the structural integrity of the assembly.

The HI-STORM 100 System is designed and evaluated for a maximum deceleration of 45g's. Studies of the capability of spent fuel rods to resist impact loads [3.5.1] indicate that the most vulnerable fuel can withstand 63 g's in the side impact orientation. Therefore, limiting the HI-STORM 100 System to a maximum deceleration of 45 g's (perpendicular to the longitudinal axis of the overpack during all normal and hypothetical accident conditions) ensures that fuel rod cladding integrity is maintained. In [3.5.1], it is assumed that the fuel rod cladding provides the only structural resistance to bending and buckling of the rod. For accidents where the predominate deceleration is directed along the longitudinal axis of the overpack, [3.5.1] also demonstrates that no elastic instability or yielding of the cladding will occur until the deceleration level is well above the HI-STORM 100 limit of 45g's. The solutions presented in [3.5.1], however, assume that the fuel pellets are not intimately attached to the cladding when subjected to an axial deceleration load that may cause an elastic instability of the fuel rod cladding.

The limit based on classical Euler buckling analyses performed by Lawrence Livermore National Laboratory in [3.5.1] is 82 g's. In the LLNL report, the limiting axial load to ensure fuel rod stability is obtained by modeling the fuel rod as a simply supported beam with unsupported length equal to the grid strap spacing. The limit load under this condition is:

$$F = \pi^2 EI/L^2$$

In the preceding formula, E = Young's Modulus of the cladding, I = area moment of inertia of the cladding, and L = spacing of the grid straps.

Assuming that  $F = W \times A/g$  with W being the weight of a fuel rod, and A = the deceleration, the Euler buckling formula can be expressed as

$$A/g = \pi^2 (ER^3 t_n / W_{\text{R}} L^2) = \pi^2 \beta$$

In the preceding formula,  $g$  = gravity,  $n$  = number of fuel rods in the fuel assembly,  $W_{fa}$  = the total weight of the fuel assembly,  $t$  = cladding wall thickness, and  $R$  = cladding mean radius.

Using the preceding formula, a survey of a large variety of fuel assembly types in [3.5.1] concluded that a 17 x 17 PWR assembly resulted in the minimum value for deceleration and results in the lower bound limit of:

$$A/g = 82$$

The fuel pellet weight was omitted from the analysis in [3.5.1] by virtue of the assumption that under axial load, the cladding did not support the fuel pellet mass. Since the results may not be conservative because of the assumption concerning the behavior of the fuel pellet mass, a new analysis of the structural response of the fuel cladding is presented here. It is demonstrated that the maximum axially oriented deceleration that can be applied to the fuel cladding is in excess of the design basis deceleration specified in this FSAR. Therefore, the initial confinement boundary remains intact during a hypothetical accident of transport where large axially directed decelerations are experienced by the HI-STORM 100 package.

The analysis reported in this section of the FSAR considers the most limiting fuel rod in the fuel assembly. Most limiting is defined as the fuel rod that may undergo the largest bending (lateral) deformations in the event of a loss of elastic stability. The fuel rod is modeled as a thin-walled elastic tube capable of undergoing large lateral displacements in the event that high axial loads cause a loss of stability (i.e., the non-linear interaction of axial and bending behavior of the elastic tube is included in the problem formulation). The fuel rod and the fuel pellet mass is included in the analysis with the fuel pellet mass assumed to contribute only its mass to the analysis. In the HI-STORM 100 spent fuel basket, continuous support to limit lateral movement is provided to the fuel assembly along its entire length. The extent of lateral movement of any fuel rod in a fuel assembly is limited to: (1) the clearance gap between the grid straps and the fuel basket cell wall at the grid strap locations; and, (2) the maximum available gap between the fuel basket cell wall and the fuel rod in the region between the grid straps. Note that the grid straps act as fuel rod spacers at the strap locations; away from the grid straps, however, there is no restraint against fuel rod-to-rod contact under a loading giving rise to large lateral motion of the individual rods. Under the incremental application of axial deceleration to the fuel rod, the fuel rod compresses and displaces from the axially oriented inertial loads experienced. The non-linear numerical analysis proceeds to track the behavior of the fuel rod up to and beyond contact with the rigid confining walls of the HI-STORM 100 fuel basket.

The analysis is carried out for the “most limiting” spent fuel assembly. The “most limiting” criteria used herein is based on the simple elastic stability formula assuming buckling occurs only between grid straps. This is identical to the methodology employed in [3.5.1] to identify the fuel assembly that limits design basis axial deceleration loading. Table 3.5.1 presents tabular data for a wide variety of fuel assemblies. Considerable data was obtained using the tables in [3.5.2]. The configuration with the lowest value of “Beta” is the most limiting for simple elastic Euler buckling between grid straps; the Westinghouse 14x14 Vantage,”W14V”, PWR configuration is used to obtain results.

The material properties used in the non-linear analysis are those for irradiated Zircalloy and are obtained from [3.5.1]. The Young's Modulus and the cladding dynamic yield stress are set as:

$$E = 10,400,000 \text{ psi}$$

$$\sigma_y = 80,500 \text{ psi}$$

The fuel cladding material is assumed to have no tensile or compressive stress capacity beyond the material yield strength.

Calculations are performed for two limiting assumptions on the magnitude of resisting moment at the grid straps. Figures 3.5.1 through 3.5.9 aid in understanding the calculation. It is shown in the detailed calculations that the maximum stress in the fuel rod cladding occurs subsequent to the cladding deflecting and contacting the fuel basket cell wall. Two limiting analyses are carried out. The initial analysis assumes that the large deflection of the cladding between two grid straps occurs without any resisting moment at the grid strap supports. This maximizes the stress in the free span of the cladding, but eliminates all cladding stress at the grid strap supports. It is shown that this analysis provides a conservative lower bound on the limiting deceleration. The second analysis assumes a reasonable level of moment resistance to develop at the grid straps; the level developed is based on an assumed deflection shape for the cladding spans adjacent to the span subject to detailed analysis. For this second analysis, the limiting decelerations are much larger with the limit stress level occurring in the free span and at the grid strap support locations.

It is concluded that the most conservative set of assumptions on structural response still lead to the conclusion that the fuel rod cladding remains intact under the design basis deceleration levels set for the HI-STORM 100.

Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA

Array ID	Array Name	Rod O.D. (in.)	Clad Thk. (in.)	R <sub>mean</sub> (in.)	# of Rods PWR	Assy Wt. (lb.)	Rod Length (in.)	# of Spans	Average Span (in.)	Material Modulus	BETA
14x14A01	W14OFA	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14A02	W14OFA	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14A03	W14V	0.4000	0.0243	0.20608	179	1177	151.85	6	25.30833	10400000	0.525127806
14x14B01	W14STD	0.4220	0.0243	0.21708	179	1302	152.4	6	25.4	10400000	0.550863067
14x14B02	XX14TR	0.4170	0.0295	0.21588	179	1215	152	6	25.33333	10400000	0.708523868
14x14B03	XX14STD	0.4240	0.0300	0.21950	179	1271.2	149.1	8	18.6375	10400000	1.337586884
14x14C01	CE14	0.4400	0.0280	0.22700	176	1270	147	8	18.375	10400000	1.398051576
14x14C02	CE14	0.4400	0.0280	0.22700	176	1220	137	8	17.125	10400000	1.67556245
14x14D01	W14SS	0.4220	0.0165	0.21513	180	1247	126.68	6	21.11333	24700000	1.31385062
15x15A01	CE15P	0.4180	0.0260	0.21550	204	1360	140	9	15.55556	10400000	1.677523904
15x15B01	W15OFA	0.4220	0.0245	0.21713	204	1459	151.85	6	25.30833	10400000	0.569346561
15x15B02	W15V5H	0.4220	0.0245	0.21713	204	1459	151.85	6	25.30833	10400000	0.569346561
15x15B03	W15	0.4220	0.0243	0.21708	204	1440	151.83	6	25.305	10400000	0.571905185
15x15B04	W15	0.4220	0.0243	0.21708	204	1443	151.83	6	25.305	10400000	0.570716193
15x15B05	15(2a-319)	0.4220	0.0242	0.21705	204	1472	151.88	6	25.31333	10400000	0.556610964
15x15C01	SPC15	0.4240	0.0300	0.21950	204	1425	152	6	25.33333	10400000	0.73601861
15x15C02	SPC15	0.4240	0.0300	0.21950	204	1425	152	6	25.33333	10400000	0.73601861
15x15C03	XX15	0.4240	0.0300	0.21950	204	1432.8	152.065	6	25.34417	10400000	0.731386148
15x15C04	XX15	0.4170	0.0300	0.21600	204	1338.6	139.423	9	15.49144	10400000	1.996693327
15x15D01	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15D02	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15D03	BW15	0.4300	0.0265	0.22163	208	1515	153.68	7	21.95429	10400000	0.854569793
15x15G01	HN15SS	0.4220	0.0165	0.21513	204	1421	126.72	6	21.12	24700000	1.305875606
16x16A01	CE16	0.3820	0.0250	0.19725	236	1430	161	10	16.1	10400000	1.270423729

Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA (continued)

Array ID	Array Name	Rod O.D. (in.)	Clad Thk. (in.)	R <sub>mean</sub> (in.)	# of Rods	Assy Wt. (lb)	Rod Length (in.)	# of Spans	Average Span (in.)	Material Modulus	BETA
16x16A02	CE16	0.3820	0.0250	0.19725	236	1300	146.499	9	16.27767	10400000	1.367126598
17x17A01	W17OFA	0.3600	0.0225	0.18563	264	1373	151.635	7	21.66214	10400000	0.613275783
17x17A02	W17OFA	0.3600	0.0225	0.18563	264	1365	152.3	7	21.75714	10400000	0.611494853
17x17B01	W17STD	0.3740	0.0225	0.19263	264	1482	151.635	7	21.66214	10400000	0.634902014
17x17B02	W17P+	0.3740	0.0225	0.19263	264	1482	151.635	7	21.66214	10400000	0.634902014
17x17C01	BW17	0.3790	0.0240	0.19550	264	1505	152.688	7	21.81257	10400000	0.687604262
BWR											
6x6A02	XX/ANF6	0.5645	0.0360	0.29125	36	328.4	116.65	4	29.1625	10400000	1.192294364
6x6C01	HB6	0.5630	0.0320	0.28950	36	270	83	3	20.75	10400000	2.500527046
7x7A01	HB7	0.4860	0.0330	0.25125	49	276	83.2	3	20.8	10400000	2.233705011
7x7B01	GE-7	0.5630	0.0320	0.28950	49	682.5	159	7	19.875	10400000	1.467601583
7x7B02	GE-7	0.5630	0.0370	0.29075	49	681	164	7	20.5	10400000	1.619330439
7x7B03	GE-7	0.5630	0.0370	0.29075	49	674.4	164	7	20.5	10400000	1.635177979
7x7B04	GE-7	0.5700	0.0355	0.29388	49	600	161.1	7	20.1375	10400000	1.887049713
7x7B05	GE-7	0.5630	0.0340	0.29000	49	600	161.1	7	20.1375	10400000	1.736760659
8x8B03	GE-8	0.4930	0.0340	0.25500	63	681	164	7	20.5	10400000	1.2906798
8x8C02	GE-8R	0.4830	0.0320	0.24950	62	600	159	7	19.875	10400000	1.352138354
8x8C03	GE-8R	0.4830	0.0320	0.24950	62	600	163.71	7	20.46375	10400000	1.27545448
9x9D01	XX/ANF9	0.4240	0.0300	0.21950	79	575.3	163.84	8	18.20444	10400000	1.367212516
10x10E01	XX10SS	0.3940	0.0220	0.20250	96	376.6	89.98	4	17.996	24700000	3.551678654

Array ID, Rod OD, Clad Thk and # of Rods from Tables 6.2.1 and 6.2.2.

R<sub>mean</sub>, Average Span and THETA are Calculated.

Zircaloy Modulus from LLNL Report [2.9.1].

Stainless Steel (348H) Modulus from ASME Code, Section III, Part D.

### Table 3.5.1 FUEL ASSEMBLY DIMENSIONAL DATA (continued)

PWR Assy. Wt., Rod Len. and # of Spans (exc. as noted below) from DOE/RW-0184, Vol. 3, UC-70, -71 and -85, Dec. 1987.

Assy. Wt., Rod Len. and # of Spans for 15x15B03, 15x15B04, 15x15C01 and 15x15C02 from ORNL/TM-9591/V1-R1.

BWR Assy. Wt., Rod Len. and # of Spans (exc. as noted below) from ORNL/TM-10902.

Assy. Wt., Rod Len. and # of Spans for 6x6A02, 9x9D01 and 10x10E01 from DOE/RW-0184, Vol. 3, UC-70, -71 and -85, Dec. 1987.

Assy. Wt., Rod Len. and # of Spans for 7x7B04 and 7x7B05 from ORNL/TM-9591/V1-R1.

Assy. Wt. for 8x8C02 and 8x8C03 from ORNL/TM-9591/V1-R1.

In the following, a physical description of the structural instability problem is provided with the aid of Figures 3.5.1 to 3.5.9. A stored fuel assembly consists of a square grid of fuel rods. Each fuel rod consists of a thin-walled cylinder surrounding and containing the fuel pellets. The majority of the total weight of a fuel rod is in the fuel pellets; however, the entire structural resistance of the fuel rod to lateral and longitudinal loads is provided by the cladding. Hereinafter, the use of the words "fuel rod", "fuel rod cladding", or just "cladding" means the structural thin cylinder. The weight of the fuel pellets is conservatively assumed to be attached to the cladding for all discussions and evaluations.

Figure 3.5.1 shows a typical fuel rod in a fuel assembly. Also shown in Figure 3.5.1 are the grid straps and the surrounding walls of the spent fuel basket cell walls. The grid straps serve to maintain the fuel rods in a square array at a certain number of locations along the length of the fuel assembly. When the fuel rod is subject to a loading causing a lateral deformation, the grid strap locations are the first locations along the length of the rod where contact with the fuel basket cell walls occurs. The fuel basket cell walls are assumed to be rigid surfaces. The fuel rod is assumed subject to some axial load and most likely has some slight initially deformed shape. For the purposes of the analysis, it is assumed that displacement under load occurs in a 2-D plane and that the ends of the fuel rod cladding have a specified boundary condition to restrain lateral deflection. The ends of the fuel rod cladding are assumed to be simply supported and the grid straps along the length of the fuel assembly are assumed to have gap " $g_1$ " relative to the cell walls of the fuel basket. The figure shows a typical fuel rod in the assembly that is located by gaps " $g_2$ " and " $g_3$ " with respect to the fuel basket walls. Because the individual fuel rod is long and slender and is not perfectly straight, it will deform under a small axial load into the position shown in Figure 3.5.2. The actual axial load is due to the distributed weight subject to a deceleration from a hypothetical accident of transport. For the purposes of this discussion, it is assumed that some equivalent axial load is applied to one end of the fuel rod cladding. Because of the distributed weight and the fact that a deceleration load is not likely to be exactly axially oriented, the predominately axial load will induce a lateral displacement of the fuel rod cladding between the two end supports. The displacement will not be symmetric but will be larger toward the end of the cladding where support against the axial deceleration is provided. Depending on the number of grid straps, either one or two grid straps will initially make contact with the fuel basket cell wall and the contact will not be exactly centered along the length of the cell. Figure 3.5.3 illustrates the position of the fuel rod after the axial load has increased beyond the value when initial contact occurred and additional grid straps are now in contact with the cell wall. The maximum stress in the fuel rod will occur at the location of maximum curvature and will be a function of the bending moment ( $F_2 \times (g_2 - g_1)$ ).

At some load  $F_3 > F_2$ , either the limit stress in the fuel rod cladding is achieved or the rod begins to experience large lateral movements between grid plates because of the coupling between axial and lateral load and deformation. Figure 3.5.4 shows the deformation mode experienced by the fuel rod cladding caused by the onset of an instability between two grid straps that are in contact with the fuel basket cell wall.

Once the lateral displacement initiates, the rod displaces until contact with the cell wall occurs at the mid point "A" ( see Figure 3.5.5) or the cladding stress exceeds the cladding material yield strength. Depending on the particular location of the fuel rod in the fuel assembly, the highest stressed portion of the fuel rod will occur in the segment with the larger of the two gaps " $g_2$ " and " $g_3$ ". For the discussion to follow, assume that  $g_2 > g_3$ . The boundary condition at the grid strap is conservatively assumed as simply-supported so that the analysis need not consider what happens in adjacent spans between grid straps. At this point in the loading process, the maximum bending moment occurs at the contact point and has the value  $F_4 \times (g_2 - g_1)$ . Figure 3.5.5 shows the displaced configuration at the load level where initial contact occurs with the fuel cell wall. If the maximum fuel rod stress (from the bending moment and from the axial load) equals the yield stress of the fuel rod cladding, it is assumed that  $F_3 = F_4$  is the maximum axial load that can be supported. The maximum stress in the fuel rod cladding occurs at point "A" in Figure 3.5.5 since that location has the maximum bending moment. If the cladding stress is still below yield, additional load can be supported. As the load is further increased, the bending moment is decreased and replaced by reaction loads, "V", at the grid strap and the contact point. These reaction loads V are shown in Figure 3.5.7 and are normal to the cell wall surface. Figure 3.5.6 shows the configuration after the load has been further increased from the value at initial contact. There are two distinct regions that need to be considered subsequent to initial contact with the fuel basket cell wall. During the additional loading phase, the point "A" becomes two "traveling" points, A, and A'. Since the bending moment at A' and A is zero, the moment  $F_5 \times (g_2 - g_1)$  is balanced by forces V at the grid strap and at point A or A'. This is shown in Figure 3.5.7 where the unsupported length current "a" is shown with the balancing load. At this point in the process, two "failure" modes are possible for the fuel rod cladding.

The axial load that develops in the unsupported region between the grid strap and point A' causes increased deformation and stress in that segment, or,

The straight region of the rod, between A and A', begins to experience a lateral deformation away from the cell wall.



Note that in this latter scenario, the slope at A or A' remains zero so this should never govern unless the flat region becomes large. The final limit load occurs when the maximum stress in either portion of the rod exceeds the yield stress of the tube. In what follows, the most limiting fuel assembly from the array of fuel types considered is subject to detailed analysis and the limit load established. This limit axial load is considered as the product of the fuel rod weight times the deceleration. Therefore, establishing the limit load to reach cladding material yield establishes the limiting axial deceleration that can be imposed.

The preceding discussion has assumed end conditions of simple support for conservatism. The location of the fuel rod determines the actual free gap between grid straps. For example, a fuel rod furthest from the cell wall that resists lateral movement of the assembly moves to close up all of the clearances that exist between it and the resisting cell wall. The clearance between rods is the rod pitch minus the rod diameter. In a 14 x 14 assembly, there are 13 clearance gaps plus an additional clearance  $g_3$  between the nearest rod and the cell wall. Therefore, the gap  $g_2$  is given as

$$g_2 = 13(\text{pitch-diameter}) + g_3$$

Figure 3.5.9 provides an illustration of the fuel rod deformation for a case of 5 fuel rods in a column. Clearly for this case, the available lateral movement can be considerable for the "furthest" fuel rod. On the other hand, for this fuel rod, there will be considerable moment resistance at the grid strap from the adjacent section of the fuel rod. The situation is different when the rod being analyzed is assumed to be the closest to the cell wall. In this case, the clearance gap is much smaller, but the moment resistance provided by adjacent sections of the rod is reduced. For calculation purposes, we assume that a moment resistance is provided as  $M = f \times K\theta$  for the fuel rod under analysis where

$K = 3EI/L$ ,  $L =$  span between grid straps, and "f" is an assumed fraction of K

The preceding result for the rotational spring constant assumes a simple support at each end of the span with an end moment "M" applied. Classical strength of materials gives the result for the spring constant. The arbitrary assumption of a constant reduction in the spring constant is to account for undetermined interactions between axial force in the rod and the calculated spring constant. As the compressive force in the adjacent members increases, the spring constant will be reduced. On the other hand, as the adjacent span contacts its near cell wall, the spring constant increases. On balance, it should be conservative to assume a considerable reduction in the spring constant available to the span being analyzed in detail. As a further conservatism, we also use the angle  $\theta$  defined by the geometry and not include any additional elastic displacement shape. This will further reduce the value of the resisting moment at any stage of the solution. In the detailed calculations, two limiting cases are examined. To limit the analysis to a single rod, it is assumed that after "stack-up" of the rods (see Figure 3.5.9), the lateral support provided by the cell wall supports all of the rods. That is, the rods are considered to have non-deforming cross-section.

Numerical Analysis - Based on the tabular results in Table 3.5.1, the fuel assembly with the smallest value for the deceleration based on the classical Euler buckling formula is analyzed in detail. The following input data is specified for the limiting 14 x 14 assembly [3.5.2]:

Inside dimension of a HI-STORM 100 fuel basket cell	$s := 8.75 \cdot \text{in}$
Outside envelope dimension of grid plate	$gp := 7.763 \cdot \text{in}$
Outer diameter of fuel rod cladding	$D := .4 \cdot \text{in}$
Wall thickness of cladding	$t := .0243 \cdot \text{in}$
Weight of fuel assembly(including end fittings)	$W := 1177 \cdot \text{lbf}$
Number of fuel rods + guide/instrument tubes in a column or row	$n := 14$
Overall length of fuel rod between assumed end support	$L_t := 151 \cdot \text{in}$
Length of fuel rod between grid straps	$L_s := 25.3 \cdot \text{in}$
Average clearance to cell wall at a grid strap location assuming a straight and centered fuel assembly	$g_1 := .5 \cdot (s - gp)$ $g_1 = 0.494 \text{ in}$
Rod pitch	$\text{pitch} := 0.556 \cdot \text{in}$
Clearance := $(n - 1) \cdot (\text{pitch} - D)$	Clearance = 2.028 in
Minimum available clearance for lateral movement of a fuel rod between grid straps	$g_3 := g_1 + .5 \cdot [gp - (n \cdot D + \text{Clearance})]$ $g_3 = 0.561 \text{ in}$
Maximum available clearances for lateral movement of a fuel rod between grid straps	$g_2 := g_3 + \text{Clearance}$ $g_2 = 2.589 \text{ in}$

Young's Modulus of Zircalloy [3.5.1]

$$E := 10400000 \cdot \text{psi}$$

Dynamic Yield Strength of Zircalloy [3.5.1]

$$\sigma_y := 80500 \cdot \text{psi}$$

Geometry Calculations:

Compute the metal cross section area A, the metal area moment of inertia I, and the total weight of a single fuel rod (conservatively assume that end fittings are only supported by fuel rods in the loading scenario of interest).

$$A := \frac{\pi}{4} \cdot [D^2 - (D - 2 \cdot t)^2]$$

$$I := \frac{\pi}{64} \cdot [D^4 - (D - 2 \cdot t)^4]$$

$$A = 0.029 \text{ in}^2$$

$$I = 5.082 \times 10^{-4} \text{ in}^4$$

$$W_r := \frac{W}{n} \quad W_r = 6.005 \text{ lbf}$$

As an initial lower bound calculation, assume no rotational support from adjacent spans and define a multiplying factor

$$f := 0.0$$

Compute the rotational spring constant available from adjacent sections of the rod.

$$K := 3 \cdot E \cdot \frac{I}{L_s} \cdot f \quad K = 0 \text{ lbf} \cdot \text{in}$$

Now compute the limit load, if applied at one end of the fuel rod cladding, that causes an overall elastic instability and contact with the cell wall. Assume buckling in a symmetric mode for a conservatively low result. The purpose of this calculation is solely to demonstrate the flexibility of the single fuel rod. No resisting moment capacity is assumed to be present at the fittings.

$$P_0 := \pi^2 \cdot E \cdot \frac{I}{L_t^2} \quad P_0 = 2.288 \text{ lbf}$$

Note that this is less than the weight of the rod itself. This demonstrates that in the absence of any additional axial support, the fuel rod will bow and be supported by the cell walls under a very small axial load. In reality, however, there is additional axial support that would increase this initial buckling load. The stress induced in the rod by this overall deflected shape is small.

$$\text{Stress}_1 := \frac{P_0 \cdot g_1 \cdot D}{2 \cdot I} \quad \text{Stress}_1 = 444.32 \text{ psi}$$

$$\text{Stress}_d := \frac{P_0}{A} \quad \text{Stress}_d = 79.76 \text{ psi}$$

The conclusion of this initial calculation is that grid straps come in contact and we need only consider what happens between a grid strap. We first calculate the classical Euler buckling load based on a pin-ended rod and assuming conservatively that the entire weight of the rod is providing the axial driving force. This gives a conservatively low estimate of the limiting deceleration that can be resisted before a perfectly straight rod buckles.

$$a_{\text{lim1}} := \pi^2 \cdot E \cdot \frac{I}{L_s^2 \cdot W_r} \quad a_{\text{lim1}} = 13.57$$

The rigid body angle of rotation at the grid strap under this load that causes contact is:

$$\theta_1 := \text{atan} \left[ 2 \cdot \frac{(g_2 - g_1)}{L_s} \right] \quad \theta_1 = 9.406 \text{ deg}$$

Conservatively assume resisting moment at the grid is proportional to this "rigid body" angle:

$$M_r := K \cdot \theta_1 \quad M_r = 0 \text{ in} \cdot \text{lbf} \quad (\text{in this first analysis, no resisting moment is assumed})$$

The total stress at the grid strap due to the axial force and the resisting moment is

$$\sigma_{\text{gs}} := \frac{W_r \cdot a_{\text{lim1}}}{A} + \frac{M_r \cdot D}{2 \cdot I} \quad \sigma_{\text{gs}} = 2841.172 \text{ psi}$$

The total stress at the contact location is

$$\text{Stress}_2 := \frac{[W_r \cdot a_{\text{lim1}} \cdot (g_2 - g_1) - M_r] \cdot D}{2 \cdot I} \quad \text{Stress}_2 = 6.721 \times 10^4 \text{ psi}$$

$$\text{Stress}_{2d} := \frac{W_r \cdot a_{\text{lim1}}}{A} \quad \text{Stress}_{2d} = 2841.172 \text{ psi}$$

$$\text{Stress}_{2t} := \text{Stress}_2 + \text{Stress}_{2d} \quad \text{Stress}_{2t} = 7.005 \times 10^4 \text{ psi}$$

This is the maximum value of the stress at this location since, for further increase in axial load, the moment will decrease with consequent large decrease in the total stress.

The safety factor is

$$\frac{\sigma_y}{\text{Stress}_{2t}} = 1.149$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{\text{ax}} := \frac{(W_r \cdot a_{\text{lim1}})}{\cos(\theta_1)} \quad P_{\text{ax}} = 82.599 \text{ lbf}$$

At this point in the load process, a certain axial load exists in the unsupported span on either side of the contact point. However, since the unsupported span is approximately 50% of the original span, the allowable deceleration limit is larger. As the axial load is incrementally increased, the moment at the contact point is reduced to zero with consequent increases in the lateral force V at the grid strap and at the contact points A and A'. Figure 3.5.8 provides the necessary information to determine the elastic deformation that occurs in the unsupported span as the axial load increases and the contact points separate (and, therefore, decreasing the free span).

From geometry, coupled with the assumption that the deflected shape is a half "sin" function with peak value "δ", the following relations are developed:

Assume "a" is a fraction of 50% of the span (the following calculations show only the final iterated assumption for the fraction

$$\varepsilon := .9 \quad a := \varepsilon \cdot \left( \frac{L_s}{2} \right) \quad a = 11.385 \text{ in}$$

Calculate "b" in Figure 3.5.8

$$b := \left[ (a)^2 + (g_2 - g_1)^2 \right]^{.5} \quad b = 11.576 \text{ in}$$

an equation for  $\delta$  can be developed from the geometric relation

$$\frac{(g_2 - g_1)}{a} := \frac{b}{2(R - \delta)} \quad \blacksquare$$

The inverse of the radius of curvature, R, at the point of peak elastic deflection of the free span, is computed as the second derivative of the assumed sin wave deflection shape. Based on the geometry in Figure 3.5.8, the peak deflection is:

$$\delta := .5 \cdot \left[ \left[ a \cdot \frac{b}{2 \cdot (g_2 - g_1)} \right]^2 + 4 \cdot \left( \frac{b}{\pi} \right)^2 \right]^{.5} - a \cdot \frac{b}{4 \cdot (g_2 - g_1)}$$

$$\delta = 0.426 \text{ in}$$

For the assumed "a", the limiting axial load capacity in the unsupported region is conservatively estimated as:

$$a_{\text{lim}2} := \pi^2 \cdot E \cdot \frac{I}{(b)^2 \cdot W_r} \quad a_{\text{lim}2} = 64.816$$

The corresponding rigid body angle is:

$$\theta_2 := \text{atan} \left[ 1 \cdot \frac{(g_2 - g_1)}{a} \right] \quad \theta_2 = 10.429 \text{ deg}$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{(W_r \cdot a_{lim2})}{\cos(\theta_2)} \quad P_{ax} = 395.763 \text{ lbf}$$

The resisting moment is

$$M_r := K \cdot \theta_2 \quad M_r = 0 \text{ in}\cdot\text{lbf}$$

The total stress in the middle of the unsupported section of free span "b" is

$$\text{stress}_3 := \frac{(P_{ax} \cdot \delta - M_r) \cdot D}{2 \cdot I} \quad \text{stress}_3 = 6.635 \times 10^4 \text{ psi}$$

$$\text{stress}_{3d} := \frac{P_{ax}}{A} \quad \text{stress}_{3d} = 1.38 \times 10^4 \text{ psi}$$

$$\text{stress}_{3t} := \text{stress}_3 + \text{stress}_{3d} \quad \text{stress}_{3t} = 8.015 \times 10^4 \text{ psi}$$

The safety factor is

$$\frac{\sigma_y}{\text{stress}_{3t}} = 1.004$$

The total stress at the grid strap due to the axial force and any the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim2}}{A} + \frac{M_r \cdot D}{2 \cdot I} \quad \sigma_{gs} = 1.357 \times 10^4 \text{ psi}$$

The safety factor is

$$\frac{\sigma_y}{\sigma_{gs}} = 5.932$$

For this set of assumptions, the stress capacity of the rod cladding has been achieved, so that the limit deceleration is:

$$A_{limit} := a_{lim2} \quad A_{limit} = 64.816$$

This exceeds the design basis for the HI-STORM 100 package.

If there is any restraining moment from the adjacent span, there is a possibility of exceeding the rod structural limits at that location due to the induced stress. Therefore, the above calculations are repeated for an assumed moment capacity at the grid strap.

$$f := 1. \quad K := 3 \cdot E \cdot \frac{I}{L_s} \cdot f$$

The rigid body angle of rotation at the grid strap under this load that causes contact is:

$$\theta_1 := \operatorname{atan} \left[ 2 \cdot \frac{(g_2 - g_1)}{L_s} \right] \quad \theta_1 = 9.406 \text{ deg}$$

Conservatively assume resisting moment at the grid a function of this angle, is

$$M_r := K \cdot \theta_1 \quad M_r = 102.875 \text{ in}\cdot\text{lbf}$$

The total stress at the grid strap due to the axial force and the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim1}}{A} + \frac{M_r \cdot D}{2 \cdot I} \quad \sigma_{gs} = 4.333 \times 10^4 \text{ psi}$$

The total stress at the contact location is

$$\text{Stress}_2 := \frac{[W_r \cdot a_{lim1} \cdot (g_2 - g_1) - M_r] \cdot D}{2 \cdot I} \quad \text{Stress}_2 = 2.672 \times 10^4 \text{ psi}$$

$$\text{Stress}_{2d} := \frac{W_r \cdot a_{lim1}}{A} \quad \text{Stress}_{2d} = 2841.172 \text{ psi}$$

$$\text{Stress}_{2t} := \text{Stress}_2 + \text{Stress}_{2d} \quad \text{Stress}_{2t} = 2.956 \times 10^4 \text{ psi}$$

This is the maximum value of the stress at this location since, for further increase in axial load, the moment will decrease with consequent large decrease in the total stress.

The axial load in the unsupported portion of the beam at this instant is

$$P_{ax} := \frac{(W_r \cdot a_{lim1})}{\cos(\theta_1)} \quad P_{ax} = 82.599 \text{ lbf}$$



At this point in the load process, a certain axial load exists in the unsupported span on either side of the contact point. However, since the unsupported span is approximately 50% of the original span, the allowable deceleration limit is larger. As the axial load is incrementally increased, the moment at the contact point is reduced to zero with consequent increases in the lateral force V at the grid strap and at the contact points A and A'. Figure 3.5.8 provides the necessary information to determine the elastic deformation that occurs in the unsupported span as the axial load increases and the contact points separate (and, therefore, decreasing the free span).

From geometry, coupled with the assumption that the deflected shape is a half "sin" function with peak value "δ", the following relations are developed:

Assume "a" is a fraction of 50% of the span (the following calculations show only the final iterated assumption for the fraction

$$\varepsilon := .7 \quad a := \varepsilon \cdot \left( \frac{L_s}{2} \right) \quad a = 8.855 \text{ in}$$

Calculate "b" in Figure 3.5.8

$$b := \left[ (a)^2 + (g_2 - g_1)^2 \right]^{.5} \quad b = 9.1 \text{ in}$$

The inverse of the radius of curvature, R, at the point of peak elastic deflection of the free span, is computed as the second derivative of the assumed sin wave deflection shape. Based on the geometry in Figure 3.5.8, the peak deflection is:

$$\delta := .5 \cdot \left[ \left[ a \cdot \frac{b}{2 \cdot (g_2 - g_1)} \right]^2 + 4 \cdot \left( \frac{b}{\pi} \right)^2 \right]^{.5} - a \cdot \frac{b}{4 \cdot (g_2 - g_1)}$$

$$\delta = 0.427 \text{ in}$$

For the assumed "a", the limiting axial load capacity in the unsupported region is conservatively estimated as:

$$a_{\text{lim}2} := \pi^2 \cdot E \cdot \frac{I}{(b)^2 \cdot W_r} \quad a_{\text{lim}2} = 104.9$$

The corresponding rigid body angle is:

$$\theta_2 := \text{atan} \left[ 1 \cdot \frac{(g_2 - g_1)}{a} \right] \quad \theta_2 = 13.314 \text{ deg}$$

The axial load in the unsupported portion of the beam at this instant is

$$P_{\text{ax}} := \frac{(W_r \cdot a_{\text{lim}2})}{\cos(\theta_2)} \quad P_{\text{ax}} = 647.331 \text{ lbf}$$

The resisting moment is

$$M_r := K \cdot \theta_2 \quad M_r = 145.619 \text{ in}\cdot\text{lbf}$$

The total stress in the middle of the unsupported section of free span "b" is

$$\text{stress}_3 := \frac{(P_{\text{ax}} \cdot \delta - M_r) \cdot D}{2 \cdot I} \quad \text{stress}_3 = 5.145 \times 10^4 \text{ psi}$$

$$\text{stress}_{3d} := \frac{P_{\text{ax}}}{A} \quad \text{stress}_{3d} = 2.257 \times 10^4 \text{ psi}$$

$$\text{stress}_{3t} := \text{stress}_3 + \text{stress}_{3d} \quad \text{stress}_{3t} = 7.402 \times 10^4 \text{ psi}$$

The safety factor is

$$\frac{\sigma_y}{\text{stress}_{3t}} = 1.088$$

The total stress at the grid strap due to the axial force and any the resisting moment is

$$\sigma_{gs} := \frac{W_r \cdot a_{lim2}}{A} + \frac{M_r \cdot D}{2 \cdot I} \quad \sigma_{gs} = 7.928 \times 10^4 \text{ psi}$$

The safety factor is  $\frac{\sigma_y}{\sigma_{gs}} = 1.015$

For this set of assumptions, the stress capacity of the rod cladding has been achieved, so that the limit deceleration is:

$$A_{limit} := a_{lim2} \quad A_{limit} = 104.9$$

### Conclusions

An analysis has demonstrated that for the most limiting PWR fuel assembly stored in the HI-STORM 100 fuel basket, a conservative lower bound limit on acceptable axial decelerations exceeds the 45g design basis of the cask. For a reasonable assumption of moment resisting capacity at the grid straps, the axial deceleration limit exceeds the design basis by a large margin.

It is concluded that fuel rod integrity is maintained in the event of a hypothetical accident condition leading to a 45g design basis deceleration in the direction normal to the target.

FUEL ROD DEFORMATION PHASES

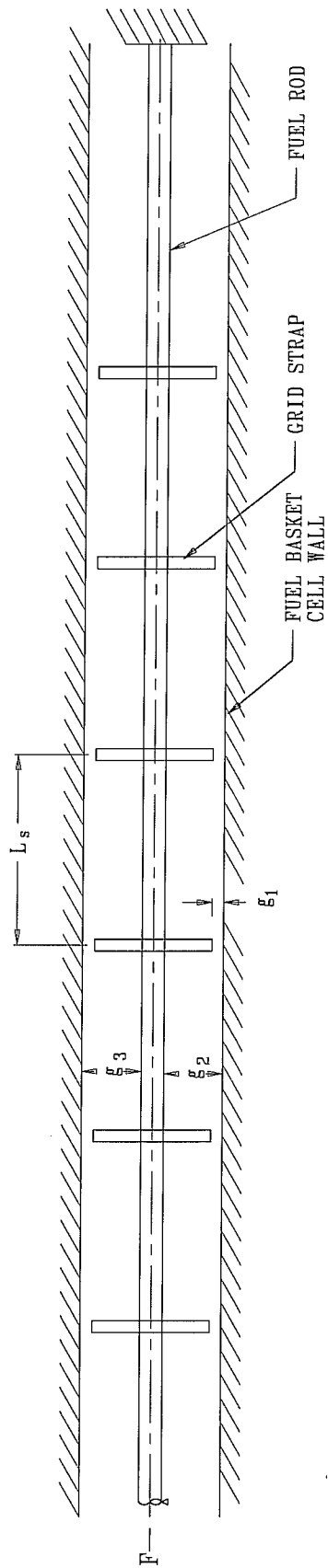


FIGURE 3.5.1;  $g_1 > 0$

FUEL ROD DEFORMATION PHASES

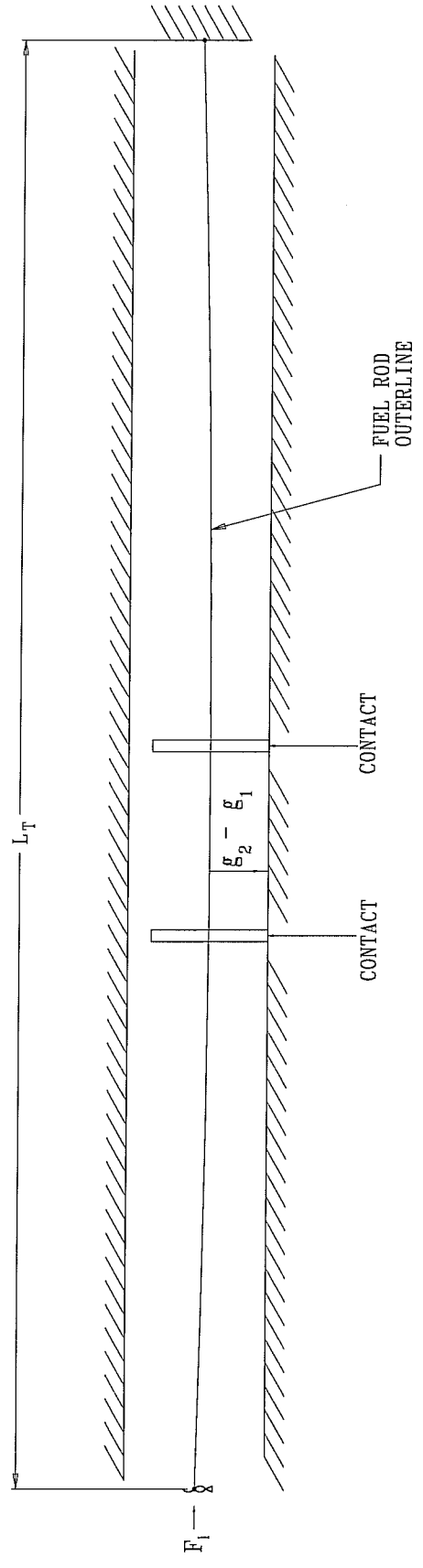


FIGURE 3.5.2:  $\epsilon_1 = 0$

FUEL ROD DEFORMATION PHASES

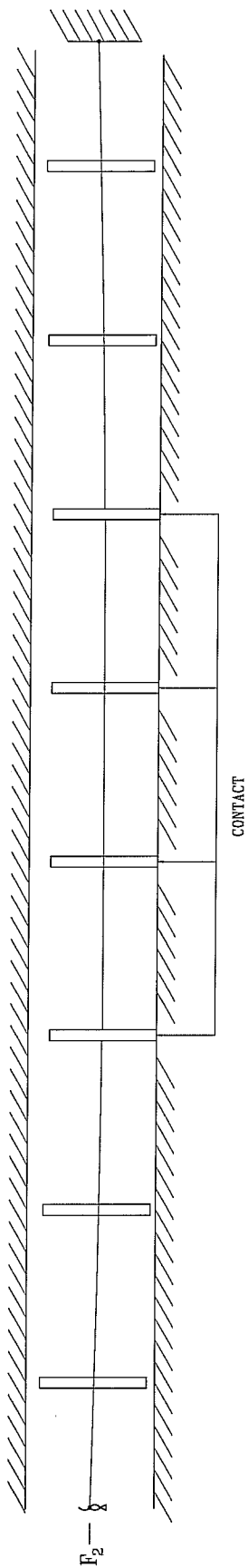


FIGURE 3.5.3:  $g_1 = 0$ ,  $F_2 > F_1$

FUEL ROD DEFORMATION PHASES

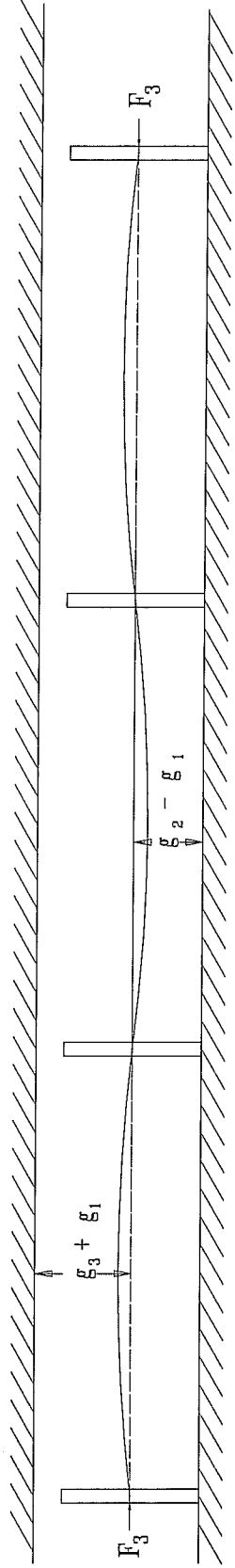


FIGURE 3.5.4; INTER-GRID STRAP DEFORMATION  $F_3 > F_2$

FUEL ROD DEFORMATION PHASES

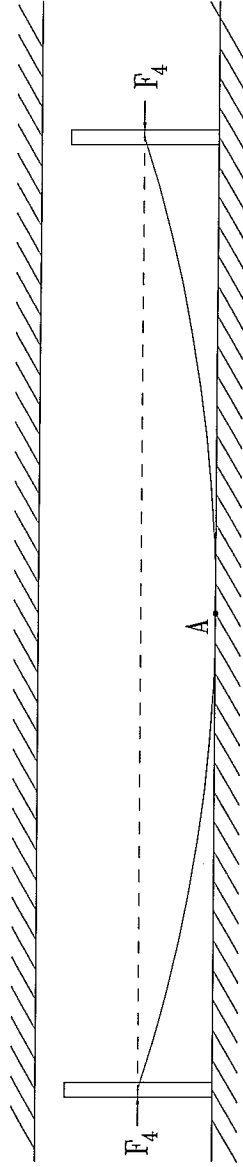


FIGURE 3.5.5; POINT CONTACT AT LOAD  $F_4$   
MAXIMUM BENDING MOMENT AT A



FUEL ROD DEFORMATION PHASES

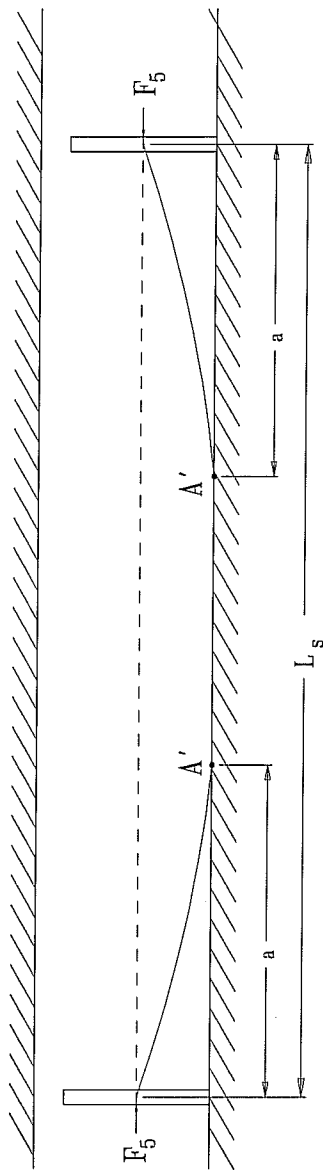


FIGURE 3.5.6.6; EXTENDED REGION OF CONTACT  
 $F_5 > F_4$ , ZERO BENDING MOMENT AT  $A'$

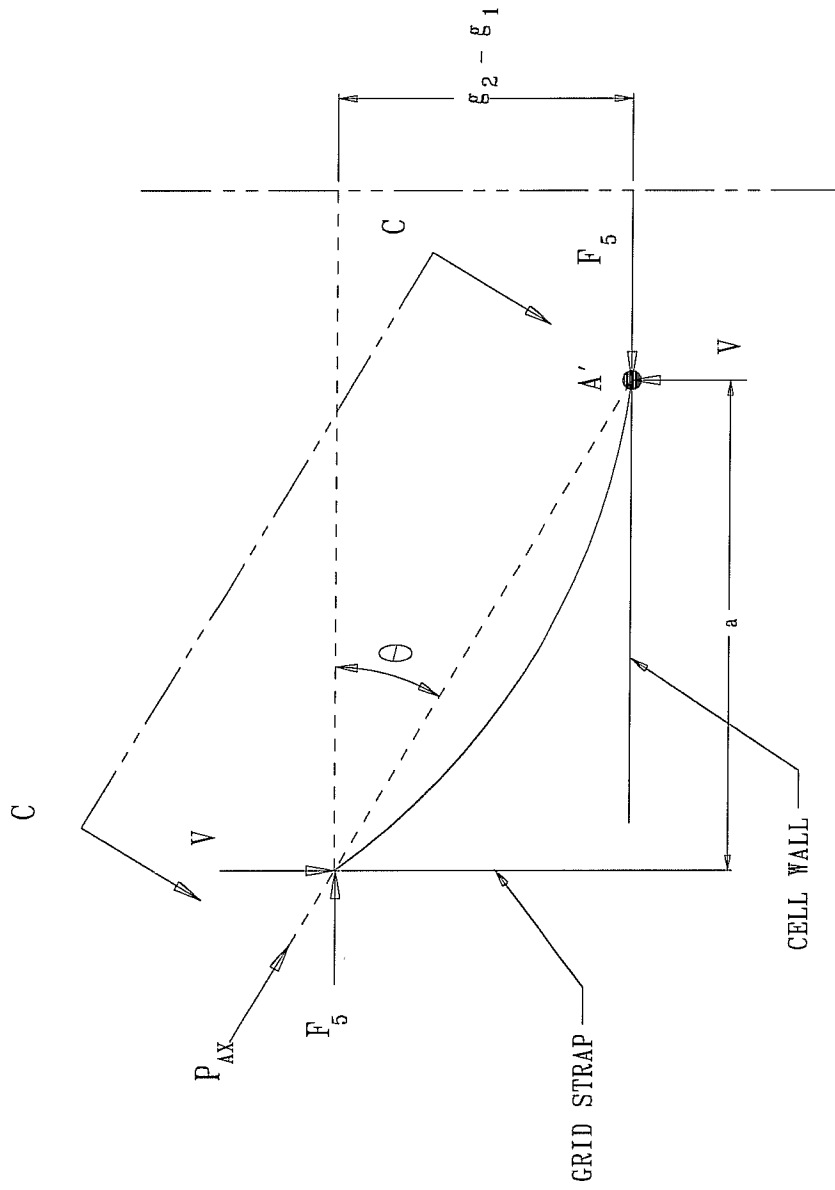
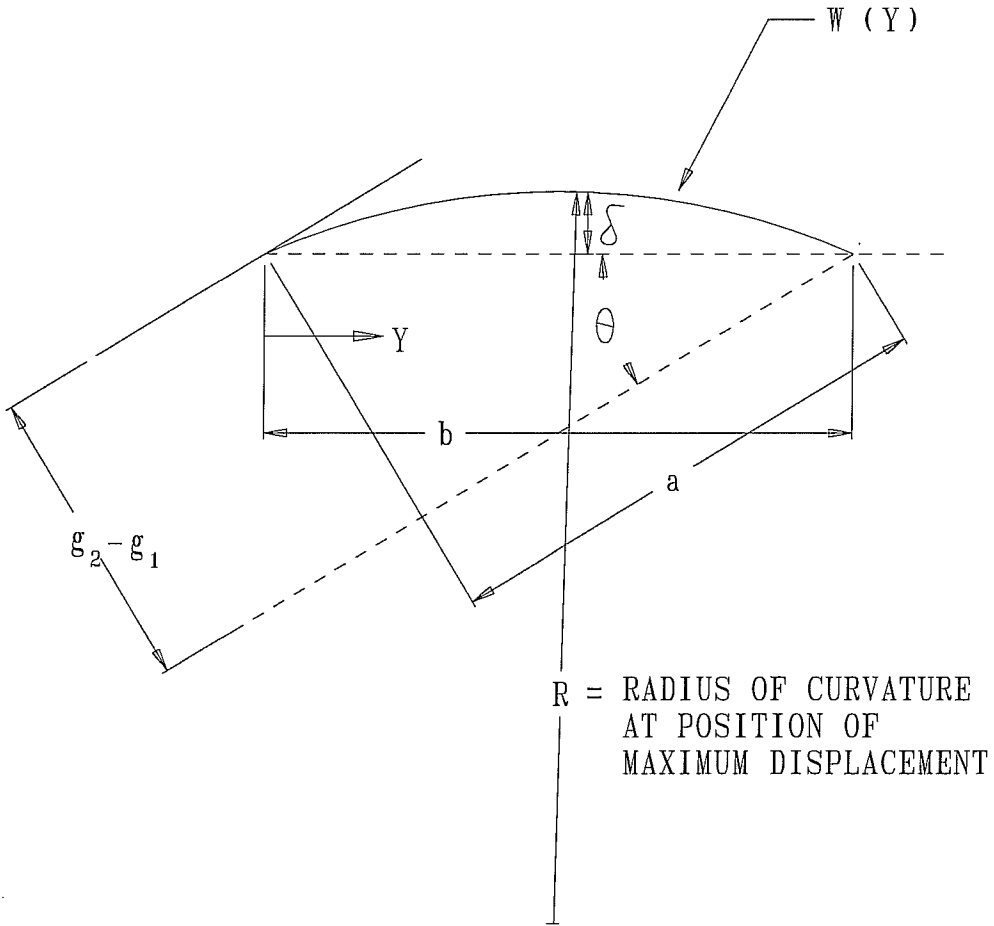


FIGURE 3.5.7; FREE BODY DIAGRAM WHEN MOMENT AT A' = 0  
 $P_{AX} = F_5 / \cos(\theta)$ . RESISTING MOMENT  $M_R$   
 AT GRID STRAP NOT SHOWN



$R =$  RADIUS OF CURVATURE  
AT POSITION OF  
MAXIMUM DISPLACEMENT

$$Z = R - \delta$$

$$W(Y) = \delta \sin(\pi Y/b)$$

FIGURE 3.5.8; VIEW C - C

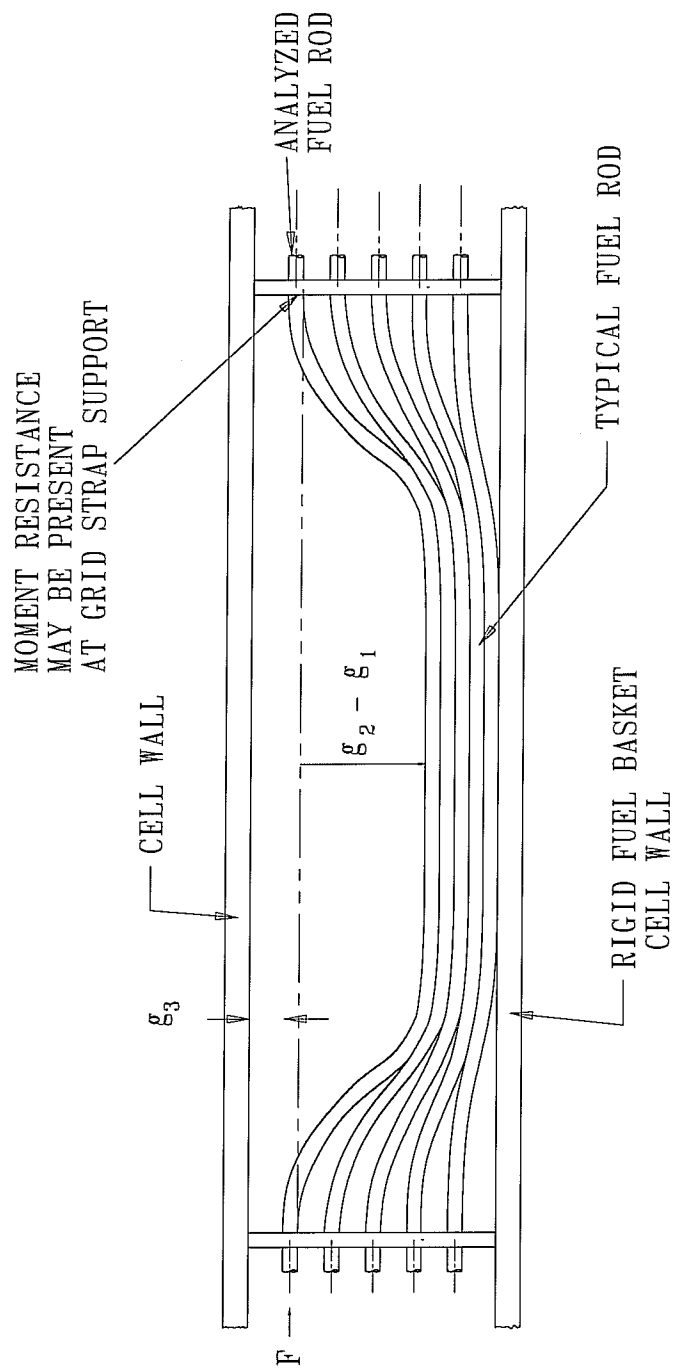


FIGURE 3.5.9; EXAGGERATED DETAIL SHOWING MULTIPLE FUEL RODS SUBJECT TO LATERAL DEFLECTION WITH FINAL STACKING OF ROD COLUMN

### 3.6 SUPPLEMENTAL DATA

#### 3.6.1 Additional Codes and Standards Referenced in HI-STORM 100 System Design and Fabrication

The following additional codes, standards and practices were used as aids in developing the design, manufacturing, quality control and testing methods for HI-STORM 100 System:

##### a. Design Codes

- (1) AISC Manual of Steel Construction, 1964 Edition and later.
- (2) ANSI N210-1976, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations".
- (3) American Concrete Institute Building Code Requirements for Structural Concrete, ACI-318-95.
- (4) Code Requirements for Nuclear Safety Related Concrete Structures, ACI349-85/ACI349R-85, and ACI349.1R-80.
- (5) ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities.
- (6) ASME NQA-2-1989, Quality Assurance Requirements for Nuclear Facility Applications.
- (7) ANSI Y14.5M, Dimensioning and Tolerancing for Engineering Drawings and Related Documentation Practices.
- (8) ACI Detailing Manual - 1980.
- (9) Crane Manufacturer's Association of America, Inc., CMAA Specification #70, Specifications for Electric Overhead Traveling Cranes, Revised 1988.

##### b. Material Codes - Standards of ASTM

- (1) E165 - Standard Methods for Liquid Penetrant Inspection.
- (2) A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.
- (3) A262 - Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.

- (4) A276 - Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.
  - (5) A479 - Steel Bars for Boilers & Pressure Vessels.
  - (6) ASTM A564, Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes.
  - (7) C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.
  - (8) A380 - Recommended Practice for Descaling, Cleaning and Marking Stainless Steel Parts and Equipment.
  - (9) C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.
  - (10) ASTM E3, Preparation of Metallographic Specimens.
  - (11) ASTM E190, Guided Bend Test for Ductility of Welds.
  - (12) NCA3800 - Metallic Material Manufacturer's and Material Supplier's Quality System Program.
- c. Welding Codes: ASME Boiler and Pressure Vessel Code, Section IX - Welding and Brazing Qualifications, 1995 Edition.
- d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling Requirements
- (1) ANSI 45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants.
  - (2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase).
  - (3) ANSI - N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants (Regulatory Guide 1.58).
  - (4) ANSI-N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants.

- (5) ANSI - N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants.
- (6) ANSI-N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants.
- (7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Equipment Materials and Services for Nuclear Power Plants (Regulatory Guide 1.123).
- (8) ANSI N45.2.15-18 - Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants.
- (9) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Regulatory Guide 1.146).
- (10) ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1995 Edition.
- (11) ANSI - N16.9-75 Validation of Calculation Methods for Nuclear Criticality Safety.

e. Reference NRC Design Documents

- (1) NUREG-0800, Radiological Consequences of Fuel Handling Accidents.
- (2) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.
- (3) NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, January 1997, Final Report.

f. Other ANSI Standards (not listed in the preceding)

- (1) ANSI/ANS 8.1 (N16.1) - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.
- (2) ANSI/ANS 8.17, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.
- (3) N45.2 - Quality Assurance Program Requirements for Nuclear Facilities - 1971.
- (4) N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.

- (5) N45.2.10 - Quality Assurance Terms and Definitions - 1973.
- (6) ANSI/ANS 57.2 (N210) - Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.
- (7) N14.6 (1993) - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials.
- (8) ANSI/ASME N626-3, Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code Section III, Div. 1, Certifying Activities.

g. Code of Federal Regulations

- (1) 10CFR20 - Standards for Protection Against Radiation.
- (2) 10CFR21 - Reporting of Defects and Non-compliance.
- (3) 10CFR50 - Appendix A - General Design Criteria for Nuclear Power Plants.
- (4) 10CFR50 - Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.
- (5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Material.
- (6) 10CFR71 - Packaging and Transportation of Radioactive Material.

h. Regulatory Guides

- (1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).
- (2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.
- (3) RG 1.28 - (ANSI N45.2) - Quality Assurance Program Requirements.
- (4) RG 1.29 - Seismic Design Classification (Rev. 3).
- (5) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Material.



- (6) RG 1.38 - (ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants.
- (7) RG 1.44 - Control of the Use of Sensitized Stainless Steel.
- (8) RG 1.58 - (ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel.
- (9) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.
- (10) RG 1.64 - (ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants.
- (11) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility.
- (12) RG 1.74 - (ANSI N45.2.10) Quality Assurance Terms and Definitions.
- (13) RG 1.85 - Materials Code Case Acceptability - ASME Section 3, Div. 1.
- (14) RG 1.88 - (ANSI N45.2.9) Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records.
- (15) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis.
- (16) RG 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.
- (17) RG 1.123 - (ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.
- (18) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, 1978.
- (19) Reg. Guide 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.
- (20) RG 3.41 - Validation of Computational Methods for Nuclear Criticality Safety, Revision 1, 1977.

- (21) Reg. Guide 8.8 - Information Relative to Ensuring that Occupational Radiation Exposure at Nuclear Power Plants will be as Low as Reasonably Achievable (ALARA).
- (22) DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants".

i. Branch Technical Position

- (1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.
- (2) ASB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

j. Standard Review Plan (NUREG-0800)

- (1) SRP 3.2.1 - Seismic Classification.
- (2) SRP 3.2.2 - System Quality Group Classification.
- (3) SRP 3.7.1 - Seismic Design Parameters.
- (4) SRP 3.7.2 - Seismic System Analysis.
- (5) SRP 3.7.3 - Seismic Subsystem Analysis.
- (6) SRP 3.8.4 - Other Seismic Category I Structures (including Appendix D), Technical Position on Spent Fuel Rack.
- (7) SRP 3.8.5 - Foundations
- (8) SRP 9.1.2 - Spent Fuel Storage, Revision 3, 1981.
- (9) SRP 9.1.3 - Spent Fuel Pool Cooling and Cleanup System.
- (10) SRP 9.1.4 - Light Load Handling System.
- (11) SRP 9.1.5 - Overhead Heavy Load Handling System.
- (12) SRP 15.7.4 - Radiological Consequences of Fuel Handling Accidents.

k. AWS Standards

- (1) AWS D1.1 - Structural Welding Code, Steel.

- (2) AWS A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.
- (3) AWS A3.0 - Standard Welding Terms and Definitions.
- (4) AWS A5.12 - Tungsten Arc-welding Electrodes.
- (5) AWS QC1 - Standards and Guide for Qualification and Certification of Welding Inspectors.

1. Others

- (1) ASNT-TC-1A - Recommended Practice for Nondestructive Personnel Qualification and Certification.
- (2) SSPC SP-2 - Surface Preparation Specification No. 2 Hand Tool Cleaning.
- (3) SSPC SP-3 - Surface Preparation Specification No. 3 Power Tool Cleaning.
- (4) SSPC SP-10 - Near-White Blast Cleaning.

3.6.2 Computer Programs

Three computer programs, all with a well established history of usage in the nuclear industry, have been utilized to perform structural and mechanical analyses documented in this report. These codes are ANSYS, DYNA3D, and WORKING MODEL. ANSYS is a public domain code which utilizes the finite element method for structural analyses.

WORKING MODEL, Version V.3.0/V.4.0

This code is used in this 10CFR72 submittal to compute the dynamic load resulting from intermediate missile impact on the overpack closure in Appendix 3.G and to evaluate the maximum elastic spring rate associated with the target during a HI-TRAC handling accident event.

WORKING MODEL has been previously utilized in similar dynamic analyses of the HI-STAR 100 system (Docket No. 72-1008).

"WORKING MODEL" (V3.0/V4.0) is a Computer Aided Engineering (CAE) tool with an integrated user interface that merges modeling, simulation, viewing, and measuring. The program includes a dynamics algorithm that provides automatic collision and contact handling, including detection, response, restitution, and friction.

Numerical integration is performed using the Kutta-Merson integrator which offers options for variable or fixed time-step and error bounding.

The Working Model Code is commercially available. Holtec has performed independent QA validation of the code (in accordance with Holtec's QA requirements) by comparing the solution of several classical dynamics problems with the numerical results predicted by Working Model. Agreement in all cases is excellent.

Additional theoretical material is available in the manual: "Users Manual, Working Model, Version 3", Knowledge Revolution, 66 Bovet Road, Suite 200, San Mateo, CA, 94402.

### DYNA3D

"DYNA3D" is a nonlinear, explicit, three-dimensional finite element code for solid and structural mechanics. It was originally developed at Lawrence Livermore Laboratories and is ideally suited for study of short-time duration, highly nonlinear impact problems in solid mechanics. DYNA3D is commercially available for both UNIX work stations and Pentium class PCs running Windows 95 or Windows NT. The PC version has been fully validated at Holtec following Holtec's QA procedures for commercial computer codes. This code is used to analyze the drop accidents and the tip-over scenario for the HI-STORM 100. Benchmarking of DYNA3D for these storage analyses is discussed and documented in Appendix 3.A.

### 3.6.3 ~~Appendices~~ Appendix Included in Chapter 3

- 3.A HI-STORM Deceleration Under Postulated Vertical Drop Event and Tipover
- ~~3.B HI-STORM 100 Overpack Deformation in Non-Mechanistic Tipover Event~~
- ~~3.C Response of Cask to Tornado Wind Load and Large Missile Impact~~
- ~~3.D Vertical Handling of Overpack with Heaviest MPC~~
- ~~3.E Lifting Trunnion Stress Analysis for HI-TRAC~~
- ~~3.F Lead Slump Analysis (HI-TRAC Side Drop)~~
- ~~3.G Missile Penetration Analysis for HI-STORM 100~~
- ~~3.H Missile Penetration Analysis for HI-TRAC~~
- ~~3.I HI-TRAC Free Thermal Expansions~~
- ~~3.J Deleted~~
- ~~3.K HI-STORM Tipover Lid Analysis~~
- ~~3.L HI-STORM Lid Top Plate Bolting~~
- ~~3.M Vertical Drop of Overpack~~
- ~~3.N Deleted.~~
- ~~3.O Deleted.~~
- ~~3.P Deleted.~~
- ~~3.Q Deleted.~~
- ~~3.R Deleted.~~
- ~~3.S Deleted.~~
- ~~3.T Deleted.~~

- ~~3.U HI STORM 100 Component Thermal Expansions MPC 24~~
- ~~3.V HI STORM 100 Component Thermal Expansions MPC 32~~
- ~~3.W HI STORM 100 Component Thermal Expansions MPC 68~~
- ~~3.X Calculation of Dynamic Load Factors~~
- ~~3.Y Miscellaneous Calculations~~
- ~~3.Z HI TRAC Horizontal Drop Analysis~~
- ~~3.AA HI TRAC 125 Rotation Trunnion Weld Analysis~~
- ~~3.AB HI TRAC Pool Lid Stress and Closure Analysis~~
- ~~3.AC Lifting Calculations~~
- ~~3.AD 125 Ton HI TRAC Transfer Lid Stress Analysis~~
- ~~3.AE Global Analysis of HI TRAC Lift~~
- ~~3.AF MPC Transfer from HI TRAC to HI STORM 100 Under Cold Conditions of Storage~~
- ~~3.AG Stress Analysis of the HI TRAC Water Jacket~~
- ~~3.AH HI TRAC Top Lid Separation Analyses~~
- ~~3.AI HI TRAC 100 Rotation Trunnion Weld Analysis~~
- ~~3.AJ 100 Ton HI TRAC Transfer Lid Stress Analysis~~
- ~~3.AK Code Case N-284 Stability Calculations~~
- ~~3.AL HI TRAC Lumped Parameters for Side Drop Analysis~~
- ~~3.AM HI TRAC 100 Transfer Cask Circumferential Deformation and Stress~~
- ~~3.AN DYNA3D Analyses of HI TRAC Side Drops and Impact by a Large Tornado Missile~~
- ~~3.AO Not used.~~
- ~~3.AP Not used.~~
- ~~3.AQ HI STORM 100 Component Thermal Expansions MPC 24E~~
- ~~3.AR Analysis of Transnuclear Damaged Fuel Canister and Thoria Rod Canister~~
- ~~3.AS Analysis of Generic PWR and BWR Damaged Fuel Containers~~

#### 3.6.4 Calculation Packages

In addition to the calculations presented in Chapter 3 and the Appendices, supporting calculation packages have been prepared to document other information pertinent to the analyses.

The calculation packages contain additional details on component weights, supporting calculations for some results summarized in the chapter, and miscellaneous supporting data that supplements the results summarized in Chapter 3 of the FSAR. All of the finite element tabular data, node and element data, supporting figures, and numerical output for all fuel baskets are contained in the calculation package supplement supporting Revision 1 of the FSAR.

### 3.7 COMPLIANCE WITH NUREG-1536

Supporting information to provide reasonable assurance with respect to the adequacy of the HI-STORM 100 System to store spent nuclear fuel in accordance with the stipulations of the Technical Specifications (Chapter 12) is provided throughout this Topical Safety Analysis Report. An itemized table (Table 3.0.1 at the beginning of this chapter) has been provided to locate and collate the substantiating material to support the technical evaluation findings listed in NUREG-1536 Chapter 3, Article VI.

The following statements are germane to an affirmative safety evaluation:

- The design and structural analysis of the HI-STORM 100 System is in full compliance with the provisions of Chapter 3 of NUREG-1536 except as listed in the Table 1.0.3 (list of code compliance exceptions).
- The list of Regulatory Guides, Codes, and standards presented in Section 3.6 herein is in full compliance with the provisions of NUREG-1536.
- All HI-STORM 100 structures, systems, and components (SSC) that are important to safety (ITS) are identified in Table 2.2.6. Section 1.5 contains the design drawings that describe the HI-STORM 100 SSCs in complete detail. Explanatory narrations in Subsections 3.4.3 *and*; 3.4.4, ~~and Chapter 3 appendices~~ provide sufficient textual details to allow an independent evaluation of their structural effectiveness.
- The requirements of 10CFR72.24 with regard to information pertinent to structural evaluation is provided in Chapters 2, 3, and 11.
- Technical Specifications pertaining to the structures of the HI-STORM 100 System have been provided in Section 12.3 herein pursuant to the requirements of 10CFR72.26.
- A series of analyses to demonstrate compliance with the requirements of 10CFR72.122(b) and (c), and 10CFR72.24(c)(3) have been performed which show that SSCs designated as ITS possess an adequate margin of safety with respect to all load combinations applicable to normal, off-normal, accident, and natural phenomenon events. In particular, the following information is provided:
  - i. Load combinations for the fuel basket, enclosure vessel, and the HI-STORM 100/HI-TRAC overpacks for normal, off-normal, accident, and natural phenomenon events are compiled in Tables 2.2.14, 3.1.1, and 3.1.3 through 3.1.5, respectively.

- ii. Stress limits applicable to the materials are found in Subsection 3.3.
- iii. Stresses at various locations in the fuel basket, the enclosure vessel, and the HI-STORM 100/HI-TRAC overpacks have been computed by analysis.

Descriptions of stress analyses are presented in Sections 3.4.3 and 3.4.4, ~~which are further elaborated in a series of appendices listed at the end of this chapter.~~

- iv. Factors of safety in the components of the HI-STORM 100 System are reported as below:

- |    |                                   |                                       |
|----|-----------------------------------|---------------------------------------|
| a. | Fuel basket                       | Tables 3.4.3 and 3.4.6                |
| b. | Enclosure vessel                  | Tables 3.4.4, 3.4.6, 3.4.7, and 3.4.8 |
| c. | HI-STORM 100 overpack/<br>HI-TRAC | Table 3.4.5                           |
| d. | Miscellaneous<br>components       | Table 3.4.9                           |
| e. | Lifting devices                   | Subsection 3.4.3                      |

- The structural design and fabrication details of the fuel baskets whose safety function in the HI-STORM 100 System is to maintain nuclear criticality safety, have been carried out to comply with the provisions of Subsection NG of the ASME Code (loc. cit.) Section III. The structural factors of safety, summarized in Tables 3.4.3 and 3.4.6 for all credible load combinations under normal, off-normal, accident, and natural phenomenon events demonstrate that the Code limits are satisfied in all cases. As the stress analyses have been performed using linear elastic methods and the computed stresses are well within the respective ASME Code limits, it follows that the physical geometry of the fuel basket will not be altered under any load combination to create a condition adverse to criticality safety. This conclusion satisfies the requirement of 10CFR72.124(a), with respect to structural margins of safety for SSCs important to nuclear criticality safety.
- Structural margins of safety during handling, packaging, and transfer operations, mandated by the provisions of 10CFR Part 72.236(b), require that the lifting and handling devices are engineered to comply with the stipulations of ANSI N14.6, NUREG-0612, Regulatory Guide 3.61, and NUREG-1536, and that the components being handled meet the applicable ASME Code service condition

stress limits. The requirements of the governing codes for handling operations are summarized in Subsection 3.4.3 herein. A summary table of factors of safety for all ITS components under lifting and handling operations, presented in Subsection 3.4.3, shows that adequate structural margins exist in all cases.

- Consistent with the requirements of 10CFR72.236(i), the confinement boundary for the HI-STORM 100 System has been engineered to maintain confinement of radioactive materials under normal, off-normal, and postulated accident conditions. This assertion of confinement integrity is made on the strength of the following information provided in this FSAR.
  - i. The MPC Enclosure Vessel which constitutes the confinement boundary is designed and fabricated in accordance with Section III, Subsection NB (Class 1 nuclear components) of the ASME Code to the maximum extent practicable.
  - ii. The MPC lid of the MPC Enclosure Vessel is welded using a strength groove weld and is subjected to volumetric examination or multiple liquid penetrant examinations, hydrostatic testing, liquid penetrant (root and final), and leakage testing to establish a maximum confidence in weld joint integrity.
  - iii. The closure of the MPC Enclosure Vessel consists of *two* independent isolation barriers.
  - iv. The confinement boundary is constructed from stainless steel alloys with a proven history of material integrity under environmental conditions.
  - v. The load combinations for normal, off-normal, accident, and natural phenomena events have been compiled (Table 2.2.14) and applied on the MPC Enclosure Vessel (confinement boundary). The results, summarized in Tables 3.4.4 through 3.4.9, show that the factor of safety (with respect to the appropriate ASME Code limits) is greater than one in all cases. Design Basis natural phenomena events such as tornado-borne missiles (large, intermediate, or small) have also been analyzed to evaluate their potential for breaching the confinement boundary. Analyses presented in Subsection 3.4.8 and Chapter 3 appendices, and summarized in unnumbered tables in Subsection 3.4.8, show that the integrity of the confinement boundary is preserved under all design basis projectile impact scenarios.
- The information on structural design included in this FSAR complies with the requirements of 10CFR72.120 and 10CFR72.122, and can be ascertained from the information contained in Table 3.7.1.



- The provisions of features in the HI-STORM 100 structural design, listed in Table 3.7.2, demonstrate compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

Table 3.7.1

NUREG –1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Location of Supporting Information in This Document
<p>i. Design and fabrication to acceptable quality standards</p>	<p>All ITS components designed and fabricated to recognized Codes and Standards:</p> <ul style="list-style-type: none"> <li>• Basket: Subsection NG, Section III</li> <li>• Enclosure Vessel: Subsection NB, loc. cit.</li> <li>• HI-STORM 100 Structure: Subsection NF, loc. cit.</li> <li>• HI-TRAC Structure: Subsection NF, loc. cit.</li> </ul>	<p>Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7</p> <p>Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7</p> <p>Subsections 2.0.2 and 3.1.1</p> <p>Subsections 2.0.3 and 3.1.1</p>
<p>ii. Erection to acceptable quality standards</p>	<ul style="list-style-type: none"> <li>• Concrete in HI-STORM 100 meets requirements of : ACI –349(85)</li> </ul>	<p>Appendix 1.D Subsection 3.3.2</p>
<p>iii. Testing to acceptable quality standards</p>	<ul style="list-style-type: none"> <li>• All non-destructive examination of ASME Code components for provisions in the Code (see exceptions in Table 2.2.15).</li> <li>• Hydrotest of pressure vessel per the Code.</li> <li>• Testing for radiation containment per provisions of NUREG-1536</li> <li>• Concrete testing in accordance with ACI-349(85)</li> </ul>	<p>Section 9.1</p> <p>Section 9.1</p> <p>Sections 7.1 and 9.1</p> <p>Appendix 1.D</p>

Table 3.7.1

## NUREG –1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Location of Supporting Information in This Document
iv. Adequate structural protection against environmental conditions and natural phenomena.	Analyses presented in Chapter 3 demonstrate that the confinement boundary will preserve its integrity under all postulated off-normal and natural phenomena events listed in Chapters 2.	Section 2.2 Chapter 11
v. Adequate protection against fires and explosions	<ul style="list-style-type: none"> <li>• The extent of combustible (exothermic) material in the vicinity of the cask system is procedurally controlled (the sole source of hydrocarbon energy is diesel in the tow vehicle).</li> <li>• Analyses show that the heat energy released from the postulated fire accident condition surrounding the cask will not result in impairment of the confinement boundary and will not lead to structural failure of the overpack. The effect on shielding will be localized to the external surfaces directly exposed to the fire which will result in a loss of the water in the water jacket for the HI-TRAC, and no significant change in the HI-STORM 100 overpack.</li> <li>• Explosion effects are shown to be bounded by the Code external pressure design basis and there is no adverse effect on ready retrievability of the MPC.</li> </ul>	Subsections 12.3.20 and 12.3.21  Subsection 11.2.4  Subsection 11.2.11 and Subsection 3.1.2.1.1.4; 3.4.7
vi. Appropriate inspection, maintenance, and testing	Inspection, maintenance, and testing requirements set forth in this FSAR are in full compliance with the governing regulations and established industry practice.	Sections 9.1 and 9.2 Chapter 12
vii. Adequate accessibility in emergencies.	<p>The HI-STORM 100 overpack lid can be removed to gain access to the multi-purpose canister.</p> <p>The HI-TRAC transfer cask has removable bottom and top lids.</p>	Chapter 8  Chapter 8

Table 3.7.1

NUREG –1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Location of Supporting Information in This Document
<p>viii. A confinement barrier that acceptably protects the spent fuel cladding during storage.</p>	<p>The peak temperature of the fuel cladding at design basis heat duty of each MPC has been demonstrated to be maintained below the limits <i>specified in ISG-11 [4.1.4]</i><del>recommended in the reports of national laboratories.</del></p> <p>The confinement barriers consist of highly ductile stainless steel alloys. The multi-purpose canister is housed in the overpack, built from a steel structure whose materials are selected and examined to maintain protection against brittle fracture under off-normal ambient (cold) temperatures (minimum of -40°F).</p>	<p>Subsection 4.4.2</p> <p>Subsection 3.1.1 Subsection 3.1.2.3</p>
<p>ix. The structures are compatible with the appropriate monitoring systems.</p>	<p>The HI-STORM 100 overpack is a thick, upright cylindrical structure with large ventilation openings near the top and bottom. These openings are designed to prevent radiation streaming while enabling complete access to temperature monitoring probes.</p>	<p>Section 1.5, Subsection 2.3.3.2</p>

Table 3.7.1

NUREG –1536 COMPLIANCE MATRIX FOR 10CFR72.120 AND 10CFR72.122 REQUIREMENTS

Item	Compliance	Location of Supporting Information in This Document
<p>x. Structural designs that are compatible with ready retrievability of fuel.</p>	<p>The fuel basket is designed to be an extremely stiff honeycomb structure such that the storage cavity dimensions will remain unchanged under all postulated normal and accident events. Therefore, the retrievability of the spent nuclear fuel from the basket will not be jeopardized.</p> <p>The MPC canister lid is attached to the shell with a groove weld which is made using an automated welding device. A similar device is available to remove the weld. Thus, access to the fuel basket can be realized.</p> <p>The storage overpack and the transfer casks are designed to withstand accident loads without suffering permanent deformations of their structures that would prevent retrievability of the MPC by normal means. It is demonstrated by analysis that there is no physical interference between the MPC and the enveloping HI-STORM storage overpack or HI-TRAC transfer cask.</p>	<p>Subsection 3.1.1</p> <p>Sections 8.1 and 8.3</p> <p>Section 3.4 and Chapter 3 Appendices</p>

Table 3.7.2

COMPLIANCE OF HI-STORM 100 SYSTEM WITH 10CFR72.236(e), ET ALS.

Item	Compliance	Location of Supporting Information in This Document
i. Redundant sealing of confinement systems.	Two physically independent lids, each separately welded to the MPC shell (Enclosure Vessel shell) provide a redundant confinement system.	Section 1.5, Drawings Section 7.1.
ii. Adequate heat removal without active cooling systems.	Thermal analyses presented in Chapter 4 show that the HI-STORM 100 System will remove the decay heat generated from the stored spent fuel by strictly passive means and maintain the system temperature within prescribed limits.	Sections 4.4 and Sections 9.1 and 9.2
iii. Storage of spent fuel for a minimum of 20 years.	The service life of the MPC, storage overpack, and HI-TRAC are engineered to be in excess of 20 years.	Subsections 3.4.11 and 3.4.12
iv. Compatibility with wet or dry spent fuel loading and unloading facilities.	<ul style="list-style-type: none"> <li>• The system is designed to eliminate any material interactions in the wet (spent fuel pool) environment.</li> <li>• The HI-TRAC transfer cask is engineered for full compatibility with the MPCs, and standard loading and unloading facilities.</li> <li>• The HI-TRAC System is engineered for MPC transfer on the ISFSI pad with full consideration of ALARA and handling equipment compatibility.</li> </ul>	<p>Subsection 3.4.1</p> <p>Subsection 8.1.1</p> <p>Subsection 8.1.1</p>

Table 3.7.2

## COMPLIANCE OF HI-STORM 100 SYSTEM WITH 10CFR72.236(e), ET ALS.

Item	Compliance	Location of Supporting Information in This Document
v. Ease of decontamination.	<ul style="list-style-type: none"> <li>• The external surface of the multi-purpose canister is protected from contamination during fuel loading through a custom designed sealing device.</li> <li>• The HI-STORM storage overpack is not exposed to contamination</li> <li>• All exposed surfaces of the HI-TRAC transfer cask are coated to aid in decontamination</li> </ul>	<p>Figures 8.1.13 and 8.1.14</p> <p>Chapter 8</p> <p>Section 1.5, Drawings</p>
vi. Inspection of defects that might reduce confinement effectiveness.	<ul style="list-style-type: none"> <li>• The MPC enclosure vessel is designed and fabricated in accordance with ASME Code, Section III, Subsection NB, to the maximum extent practical.</li> <li>• Hydrostatic testing, helium leakage testing, and NDE of the closure welds verify containment effectiveness.</li> </ul>	Section 9.1
vii. Conspicuous and durable marking.	<p>The stainless steel lid of each MPC will have model number and serial number engraved for ready identification.</p> <p>The exterior envelope of the cask (the storage overpack) is marked in a conspicuous manner as required by 10CFR 72.236(k).</p>	N/A
viii. Compatibility with removal of the stored fuel from the site, transportation, and ultimate disposal by the U.S. Department of Energy.	The MPC is designed to be in full compliance with the DOE's draft specification for transportability and disposal published under the now dormant "MPC" program.	Section 2.4 Subsection 1.2.1.1

### 3.8 REFERENCES

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- [3.4.10] ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 1995.
- [3.4.11] 10CFR71, Waste Confidence Decision Review, USNRC, September 11, 1990.
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- [3.5.2] Physical and Decay Characteristics of Commercial LWR Spent Fuel, Oak Ridge National Laboratory Report, J. Roddy, H. Claiborne, R. Ashline, P. Johnson, and B. Rhyne, ORNL/TM-9591/V1-R1, 1/86.

## APPENDIX 3.A: HI-STORM DECELERATION UNDER POSTULATED VERTICAL DROP EVENT AND TIPOVER

### 3.A.1 INTRODUCTION

Handling accidents with a HI-STORM overpack containing a loaded MPC are credible events (Section 2.2.3). The stress analyses carried out in Chapter 3 of this safety analysis report assume that the inertial loading on the load bearing members of the MPC, fuel basket, and the overpack due to a handling accident are limited by the Table 3.1.2 decelerations. The maximum deceleration experienced by a structural component is the product of the rigid body deceleration sustained by the structure and the dynamic load factor (DLF) applicable to that structural component. The dynamic load factor (DLF) is a function of the contact impulse and the structural characteristics of the component. A solution for dynamic load factors is provided in Appendix 3.X.

The rigid body deceleration is a strong function of the load-deformation characteristics of the impact interface, weight of the cask, and the drop height or angle of free rotation. For the HI-STORM 100 System, the weight of the structure and its surface compliance characteristics are known. However, the contact stiffness of the ISFSI pad (and other surfaces over which the HI-STORM 100 may be carried during its movement to the ISFSI) is site-dependent. The contact resistance of the collision interface, which is composed of the HI-STORM 100 and the impacted surface compliance, therefore, is not known a priori for a specific site. Analyses for the rigid body decelerations are, therefore, presented here using a reference ISFSI pad (which is the pad used in a recent Lawrence Livermore National Laboratory report and is the same reference pad used in the HI-STAR 100 FSAR). The finite element model (grid size, extent of model, soil properties, etc.) follows the LLNL report.

An in-depth investigation by the Lawrence Livermore Laboratory (LLNL) into the mechanics of impact between a cask-like impactor on a reinforced concrete slab founded on a soil-like subgrade has identified three key parameters, namely, the thickness of the concrete slab,  $t_p$ , compressive strength of the concrete  $f_c'$  and equivalent Young's Modulus of the subgrade  $E$ . These three parameters are key variables in establishing the stiffness of the pad under impact scenarios. The LLNL reference pad parameters, which we hereafter denote as Set A, provide one set of values of  $t_p$ ,  $f_c'$ , and  $E$  that are found to satisfy the deceleration criteria applicable to the HI-STORM 100 cask. Another set of parameters, referred to as Set B herein, is also shown to satisfy the g-load limit requirements. In fact, an infinite number of combinations of  $t_p$ ,  $f_c'$ , and  $E$  can be compiled that would meet the g-load limit qualification. However, in addition to satisfying the g-limit criterion, the pad must be demonstrated to possess sufficient flexural and shear stiffness to meet the ACI 318 strength limits under factored load combinations. The minimum strength requirement to comply with ACI 318 provisions places a restriction on the lower bound values of  $t_p$ ,  $f_c'$ , and  $E$  that must be met in an ISFSI pad design.

Our focus in this appendix, however, is to quantify the peak decelerations that would be experienced by a loaded HI-STORM 100 cask under the postulated impact scenarios for the two pad designs defined by parameter Sets A and B, respectively. The information presented in this appendix also serves to further authenticate the veracity of the Holtec DYNA3D model described in the 1997 benchmark report [3.A.4.]

### 3.A.2 Purpose

The purpose of this appendix is to demonstrate that the rigid body deceleration experienced by the HI-STORM 100 System during a handling accident or non-mechanistic tip-over are below the design basis deceleration of 45g's (Table 3.1.2). Two accidental drop scenarios of a loaded HI-STORM 100 cask on the ISFSI pad are considered in this appendix. They are:

- i. Tipover: A loaded HI-STORM 100 is assumed to undergo a non-mechanistic tipover event and impacting the ISFSI pad with an incipient impact angular velocity, which is readily calculated from elementary dynamics.
- ii. End drop: The loaded HI-STORM 100 is assumed to drop from a specified height  $h$ , with its longitudinal axis in the vertical orientation, such that its bottom plate impacts the ISFSI pad.

It is shown in Appendix 3.X that dynamic load factors are a function of the predominate natural frequency of vibration of the component for a given input load pulse shape. Dynamic load factors are applied, as necessary, to the results of specific component analyses performed using the loading from the design basis rigid body decelerations. Therefore, for the purposes of this Appendix 3.A, it is desired to demonstrate that the rigid body deceleration experienced in each of the drop scenarios is below the HI-STORM 100 45g design basis.

### 3.A.3 Background and Methodology

In 1997 Lawrence Livermore National Laboratory (LLNL) published the experimentally obtained results of the so-called fourth series billet tests [3.A.1] together with a companion report [3.A.2] documenting a numerical solution that simulated the drop test results with reasonable accuracy. Subsequently, USNRC personnel published a paper [3.A.3] affirming the NRC's endorsement of the LLNL methodology. The LLNL simulation used modeling and simulation algorithms contained within the commercial computer code DYNA3D [3.A.6].

The LLNL cask drop model is not completely set forth in the above-mentioned LLNL reports. Using the essential information provided by the LLNL [3.A.2] report, however, Holtec is able to develop a finite element model for implementation on LS-DYNA3D [3.A.5] which is fully consistent with LLNL's (including the use of the Butterworth filter for discerning rigid body deceleration from "noisy" impact data). The details of the LS-DYNA3D dynamic model, henceforth referred to as the Holtec model, are contained in the proprietary benchmark report [3.A.4] wherein it is shown that the peak deceleration in every case of billet drop analyzed by LLNL is replicated within a small tolerance by the Holtec model. The case of the so-

called "generic" cask, for which LLNL provided predicted response under side drop and tipover events, is also bounded by the Holtec model. In summary, the benchmarking effort documented in [3.A.4] is in full compliance with the guidance of the Commission [3.A.3].

Having developed and benchmarked an LLNL-consistent cask impact model, a very similar model is developed and used to prognosticate the HI-STORM drop scenarios. The reference elasto-plastic-damage characteristics of the target concrete continuum used by LLNL, and used in the HI-STAR 100 FSAR are replicated herein. The HI-STORM 100 target model is identical in all aspects to the reference pad approved for the HI-STAR 100 FSAR.

In the tipover scenario the cask surface structure must be sufficiently pliable to cushion the impact and limit the rigid body deceleration. The angular velocity at the contact time is readily calculated using planar rigid body dynamics and is used as an initial condition in the LS-DYNA3D simulation.

The end drop event produces a circular impact patch equal to the diameter of the overpack baseplate. The elasto-plastic-damage characteristics of the concrete target and the drop height determine the maximum deceleration. A maximum allowable height "h" is determined to limit the deceleration to a value below the design basis.

A description of the work effort and a summary of the results are presented in the following sections. In all cases, the reported decelerations are below the design basis of 45g's at the top of the MPC fuel basket.

### 3.A.4 Assumptions and Input Data

#### 3.A.4.1 Assumptions

The assumptions used to create the model are completely described in Reference [3.A.4] and are shown there to be consistent with the LLNL simulation. There are key aspects, however, that are restated here:

The maximum deceleration experienced by the cask during a collision event is a direct function of the structural rigidity (or conversely, compliance) of the impact surface. The compliance of the ISFSI pad is quite obviously dependent on the thickness of the pad,  $t_p$ , the compressive strength of the concrete,  $f_c'$  and stiffness of the sub-grade (expressed by its effective Young's modulus, E). The structural rigidity of the ISFSI pad will increase if any of the three above-mentioned parameters ( $t_p$ ,  $f_c'$  or E) is increased. For the reference pad, the governing parameters (i.e.,  $t_p$ ,  $f_c'$  and E) are assumed to be identical to the pad defined by LLNL [3.A.2], which is also the same as the pad utilized in the benchmark report [3.A.4]. We refer to the LLNL ISFSI pad parameters as Set A. (Table 3.A.1).

As can be seen from Table 3.A.1, the nominal compressive strength  $f_c'$  in Set A is limited to 4200 psi. However, experience has shown that ISFSI owners have considerable practical difficulty in limiting the 28 day strength of poured concrete to 4200 psi, chiefly because a principal element of progress in reinforced concrete materials technology has been in realizing ever increasing concrete nominal strength. Inasmuch as a

key objective of the ISFSI pad is to limit its structural rigidity (and not  $f_c'$  per se), and limiting  $f_c'$  to 4200 psi may be problematic in certain cases, an alternative set of reference pad parameters is defined (Set B in Table 3.A.1), which permits a higher value of  $f_c'$  but much smaller values of pad thickness,  $t_p$  and sub-grade Young's modulus, E.

The ISFSI owner has the option of constructing the pad to comply with the limits of Set A or Set B without performing site-specific cask impact analyses. It is recognized that, for a specific ISFSI site, the reinforced concrete, as well as the underlying engineered fill properties, may be different at different locations on the pad or may be uniform, but non-compliant with either Set A or Set B. In that case, the site-specific conditions must be performed to demonstrate compliance with the design limits of the HI-STORM system (e.g., maximum rigid body g-load less than 45 g's). The essential data which define the pad (Set A and Set B) used to qualify the HI-STORM 100 are provided in Table 3.A.1.

The HI-STORM 100 steel structural elements (outer shell, inner shell, radial plates, lid, etc.), are fabricated from SA-516 Grade 70. The steel is described as a bi-linear elastic-plastic material with limited strain failure by five material parameters (E,  $S_y$ ,  $S_u$ ,  $\epsilon_u$ , and  $\nu$ ). The numerical values used in the finite element model are shown in Table 3.A.2. The concrete located inside of the overpack for this dynamic analysis is defined to be identical with the concrete pad. This is conservative since the concrete assumed in the reference pad is reinforced. Therefore, the strength of the concrete inside the HI-STORM 100 absorbs less energy if it is also assumed to be reinforced.

#### 3.A.4.2 Input Data

Table 3.A.1 characterizes the properties of the full-scale reference target pad used in the analysis of the full size HI-STORM 100 System. The principal strength parameters that define the stiffness of the pad, namely,  $t_p$ , E and  $f_c'$  are input in the manner described in [3.A.2] and [3.A.4].

Table 3.A.2 contains the material description parameters for the steel types; SA-516-70 used in the numerical investigation.

Table 3.A.3 details the geometry of the HI-STORM 100 used in the drop simulations. This data is taken from applicable HI-STORM 100 drawings.

#### 3.A.5 Finite Element Model

The finite-element model of the Holtec HI-STORM 100 overpack (baseplate, shells, radial plates, lid, concrete, etc.), concrete pad and a portion of the subgrade soil is constructed using the pre-processor integrated with the LS-DYNA3D software [3.A.5]. The deformation field for all postulated drop events (the end-drop and the tipover) exhibits symmetry with the vertical plane passing through the cask diameter and the concrete pad length. Using this symmetry condition of the deformation field only a half finite-element model is constructed. The finite-element model is organized into nineteen independent parts (the baseplate components, the outer shell, the inner shell, the radial plates, the channels, the lid components, the basket

steel plates, the basket fuel zone, the concrete pad and the soil). The final model contains 30351 nodes, 24288 solid type finite-elements, 1531 shell type finite-elements, seven (7) materials, ten (10) properties and twenty-four (24) interfaces. The finite-element model used for the tipover-drop event is depicted in Figures 3.A.1 through 3.A.4. Figures 3.A.5 through 3.A.8 show the end-drop finite-element model.

The soil grid, shown in Figure 3.A.9, is a rectangular prism (800 inches long, 375 inches wide and 470 inches deep), is constructed from 13294 solid type finite-elements. The material defining this part is an elastic isotropic material. The central portion of the soil (400 inches long, 150 inches wide and 170 inches deep) where the stress concentration is expected to appear is discretized with a finer mesh.

The concrete pad is 320 inches long, 100 inches wide and is 36 inches thick. This part contains 8208 solid finite-elements. A uniform sized finite-element mesh, shown in Figure 3.A.10, is used to model the concrete pad. The concrete behavior is described using a special constitutive law and yielding surface (MAT\_PSEUDO\_TENSOR) contained within LS-DYNA3D. The geometry, the material properties, and the material behavior are identical to the LLNL reference pad (Material 16 IIB).

The half portion of the steel cylindrical overpack contains 1531 shell finite-elements. The steel material description (SA-516-70) is realized using a bi-linear elasto-plastic constitutive model (MAT\_PIECEWISE\_LINEAR\_PLASTICITY). Figure 3.A.11 depicts details of the steel components of the cask finite-element mesh, with the exception of the inner shell, channels and lid components, which are shown in Figures 3.A.12 and 3.A.13. The concrete filled between the inner and the outer shells, and contained in the baseplate and lid components is modeled using 1664 solid finite-elements and is depicted in Figure 3.A.14. The concrete material is defined identical to the pad concrete.

The MPC and the contained fuel are modeled in two parts that represent the lid and baseplate, and the fuel area. An elastic material is used for both parts. The finite-element mesh pertinent to the MPC contains 1122 solid finite-elements and is shown in Figure 3.A.15. The mass density is appropriate to match a representative weight of 356,521 lb. that is approximately mid-way between the upper and lower weight estimates for a loaded HI-STORM 100.

The total weight used in the analysis is approximately 2,000 lb. lighter than the HI-STORM 100 containing the lightest weight MPC.

Analysis of a single mass impacting a spring with a given initial velocity shows that both the maximum deceleration " $a_M$ " of the mass and the time duration of contact with the spring " $t_c$ " are related to the dropped weight " $w$ " and drop height " $h$ " as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}; t_c \sim \sqrt{w}$$

Therefore, the most conservatism is introduced into the results by using the minimum weight. It is emphasized that the finite element model described in the foregoing is identical in its approach to the "Holtec

model" described in the benchmark report [3.A.4]. Gaps between the MPC and the overpack are included in the model.

### 3.A.6 Impact Velocity

#### a. Linear Velocity: Vertical Drops

For the vertical drop event, the impact velocity,  $v$ , is readily calculated from the Newtonian formula:

$$v = \sqrt{(2gh)}$$

where

- $g =$  acceleration due to gravity
- $h =$  free-fall height

#### b. Angular Velocity: Tip-Over

The tipover event is an artificial construct wherein the HI-STORM 100 overpack is assumed to be perched on its edge with its C.G. directly over the pivot point A (Figure 3.A.16). In this orientation, the overpack begins its downward rotation with zero initial velocity. Towards the end of the tip-over, the overpack is horizontal with its downward velocity ranging from zero at the pivot point (point A) to a maximum at the farthest point of impact (point E in Figure 3.A.17). The angular velocity at the instant of impact defines the downward velocity distribution along the contact line.

In the following, an explicit expression for calculating the angular velocity of the cask at the instant when it impacts on the ISFSI pad is derived. Referring to Figure 3.A.16, let  $r$  be the length AC where C is the cask centroid. Therefore,

$$r = \left( \frac{d^2}{4} + h^2 \right)^{1/2}$$

The mass moment of inertia of the HI-STORM 100 System, considered as a rigid body, can be written about an axis through point A, as

$$I_A = I_c + \frac{W}{g} r^2$$

where  $I_c$  is the mass moment of inertia about a parallel axis through the cask centroid C and W is the weight of the cask ( $W = Mg$ ).

Let  $\theta_1(t)$  be the rotation angle between a vertical line and the line AC. The equation of motion for rotation of the cask around point A, during the time interval prior to contact with the ISFSI pad, is

$$I_A \frac{d^2 \mathbf{q}_1}{dt^2} = Mgr \sin \mathbf{q}_1$$

This equation can be rewritten in the form

$$\frac{I_A}{2} \frac{d(\dot{\mathbf{q}}_1)^2}{d \mathbf{q}_1} = Mgr \sin \mathbf{q}_1$$

which can be integrated over the limits  $\theta_1 = 0$  to  $\theta_1 = \theta_{2f}$  (See Figure 3.A.17).

The final angular velocity  $\dot{\theta}_1$  at the time instant just prior to contact with the ISFSI pad is given by the expression

$$\dot{\mathbf{q}}_1 (t_B) = \sqrt{\frac{2 Mgr}{I_A} (1 - \cos \mathbf{q}_{2f})}$$

where, from Figure 3.A.17

$$\mathbf{q}_{2f} = \cos^{-1} \left( \frac{d}{2r_1} \right)$$

This equation establishes the initial conditions for the final phase of the tip-over analysis; namely, the portion of the motion when the cask is decelerated by the resistive force at the ISFSI pad interface.

Using the data germane to HI-STORM 100 (Table 3.A.3), and the above equations, the angular velocity of impact is calculated as 1.49 rad/sec.



### 3.A.7 Results

#### 3.A.7.1 Set A Pad Parameters

It has been previously demonstrated in the benchmark report [3.A.4] that bounding rigid body decelerations are achieved if the cask is assumed to be rigid with only the target (ISFSI pad) considered as an energy absorbing media. Therefore, for the determination of the bounding decelerations reported in this appendix, the HI-STORM storage overpack was conservatively made rigid except for the radial channels that position the MPC inside of the overpack. The MPC material behavior was characterized in the identical manner used in the Livermore Laboratory analysis as was the target ISFSI pad and underlying soil. The LS-DYNA3D time-history results are processed using the Butterworth filter (in conformance with the LLNL methodology) to establish the rigid body motion time-history of the cask. The material points on the cask where the acceleration displacement and velocity are computed for each of the drop scenarios are shown in Figure 3.A.18.

Node 82533 (Channel A1), which is located at the center of the outer surface of the baseplate, serves as the reference point for end-drop scenarios.

Node 84392 (Channel A2), which is located at the center of the cask top lid outer surface, serves as the reference point for the tipover scenario with the pivot point indicated as Point 0 in Figure 3.A.18.

The final results are shown in Table 3.A.4.

i. Tipover:

The time-histories of the impact force, the displacement and velocity time-histories of Channel A2, and the average vertical deceleration of the overpack lid top plate have been determined for this event [3.A.7].

The deceleration at the top of the fuel basket is obtained by ratioing the average deceleration of the overpack lid top plate. The maximum filtered deceleration at the top of the fuel basket is 42.85g's, which is below the design basis limit.

ii. End Drop:

The drop height  $h = 11$ " is considered in the numerical analysis. This is considered as an acceptable maximum carry height for the HI-STORM 100 System if lifted above a surface with design values of  $t_p$ ,  $f_c'$ , and  $E$  equal to those presented in Table 3.A.1 for Parameter Set "A". The maximum filtered deceleration at the top of the fuel basket is 43.98g's, which is below the design basis limit.

The computer code utilized in this analysis is LS-DYNA3D [3.A.5] validated under Holtec's QA system. Table 3.A.4 summarizes the key results from all impact simulations for the Set A parameters discussed in the foregoing.

The filter frequencies (to remove unwanted high-frequency contributions) for the Holtec cask analyses analyzed in this TSAR is the same as used for the corresponding problem analyzed in [3.A.2] and [3.A.4]. To verify the Butterworth filter parameters (350 Hz cutoff frequency, etc.) used in processing the numerical data, a Fourier power decomposition was generated.

### 3.A.7.2 Set B Parameters

As stated previously, Set B parameters produce a much more compliant pad than the LLNL reference pad (Set A). This fact is borne out by the tipover and end analyses performed on the pad defined by the Set B parameters. Table 3.A.4 provides the filtered results for the two impact scenarios. In every case, the peak decelerations corresponding to Set B parameters are less than those for Set A (also provided in Table 3.A.4).

Impact force and acceleration time history curves for Set B have the same general shape as those for Set A and are contained in the calculation package [3.A.7]. All significant results are summarized in Table 3.A.4.

### 3.A.8 Computer Codes and Archival Information

The input and output files created to perform the analyses reported in this appendix are archived in Holtec International calculation package [3.A.7].

### 3.A. 9 Conclusion

The DYNA3D analysis of HI-STORM 100 reported in this appendix leads to the following conclusion:

- a. If a loaded HI-STORM undergoes a free fall for a height of 11 inches in a vertical orientation on to a reference pad defined by Table 3.A.1, the maximum rigid body deceleration is less than 45g's for both Set A and Set B pad parameters.
- b. If a loaded HI-STORM 100 overpack pivots about its bottom edge and tips over on to a reference pad defined by Table 3.A.1, then the maximum rigid body deceleration of the cask centerline at the plane of the top of the MPC fuel basket cellular region is less than 45g's for both Set A and Set B parameters..

Table 3.A.4 provides key results for all drop cases studied herein for both pad parameter sets (A and B). If the pad designer maintains each of the three significant parameters ( $t_p$ ,  $f_c'$ , and E) below the limit for the specific set selected (Set A or Set B), then the stiffness of the pad at any ISFSI site will be lower and the computed decelerations at the ISFSI site will also be lower. Furthermore, it is recognized that a refinement of the cask dynamic model will accrue further reduction in the computed peak deceleration. For example, incorporation of the structural flexibility in the MPC enclosure vessel, fuel basket, etc., would lead to additional reductions in the computed values of the peak deceleration. These refinements, however, add to the computational complexity. Because g-limits are met without the above-mentioned and other refinements in the cask dynamic model, the simplified dynamic model described in this appendix was retained to reduce the overall computational effort.

3.A.10 References

- [3.A.1] Witte, M., et al., "Evaluation of Low-Velocity Impacts Tests of Solid Steel Billet onto Concrete Pads.", Lawrence Livermore National Laboratory, UCRL-ID-126274, Livermore, California, March 1997.
- [3.A.2] Witte, M., et al., "Evaluation of Low-Velocity Impacts Tests of Solid Steel Billet onto Concrete Pads, and Application to Generic ISFSI Storage Cask for Tipover and Side Drop.", Lawrence Livermore National Laboratory, UCRL-ID-126295, Livermore, California, March 1997.
- [3.A.3] Tang, D.T., Raddatz, M.G., and Sturz, F.C., "NRC Staff Technical Approach for Spent Fuel Cask Drop and Tipover Accident Analysis", SFPO, USNRC (1997).
- [3.A.4] Simulescu, I., "Benchmarking of the Holtec LS-DYNA3D Model for Cask Drop Events", Holtec Report HI-971779, September 1997.
- [3.A.5] LS-DYNA3D, Version 936-03, Livermore Software Technology Corporation, September 1996.
- [3.A.6] Whirley, R.G., "DYNA3D, A Nonlinear, Explicit, Three-Dimensional Finite element Code for Solid and Structural Mechanics - User Manual.", Lawrence Livermore National Laboratory, UCRL-MA-107254, Revision 1, 1993.
- [3.A.7] Zhai, J. "Analysis of the Loaded HI-STORM 100 System Under Drop and Tip-Over Scenarios", Holtec Report HI-2002474, July 2000.

Table 3.A.1: Essential Variables to Characterize the ISFSI Pad (Set A and Set B)

Item	Parameter Set A	Parameter Set B
Thickness of concrete, (inches)	36	28
Nominal compressive strength of concrete at 28 days, (psi)	4,200	6,000
Max. modulus of elasticity of the subgrade (psi)	28,000	16,000

- Notes:
1. The concrete Young's Modulus is derived from the American Concrete Institute recommended formula  $57,000\sqrt{f}$  where  $f$  is the nominal compressive strength of the concrete (psi).
  2. The effective modulus of elasticity of the subgrade will be measured by the classical "plate test" or other appropriate means before pouring of the concrete to construct the ISFSI pad.
  3. The pad thickness, concrete compressive strength, and the subgrade soil effective modulus are the upper bound values to ensure that the deceleration limits under the postulated events set forth in Table 3.1.2 are satisfied.

Table 3.A.2: Essential Steel Material Properties for HI-STORM 100 Overpack

Steel Type	Parameter	Value
SA-516-70 at T = 350 deg. F	E	2.800E + 07
	S <sub>y</sub>	3.315E+04 psi
	S <sub>u</sub>	7.000E+04 psi
	ε <sub>u</sub>	0.21
	ν	0.30

Note that the properties of the steel components, except for the radial channels used to position the MPC, do not affect the results reported herein since the HI-STORM 100 is eventually assumed to behave as a rigid body (by internal constraint equations automatically computed by DYNA3D upon issue of a “make rigid” command). In Section 3.4, however, stress and strain results for an additional tip-over analysis, performed using the actual material behavior ascribed to the storage overpack, are presented for the sole purpose of demonstrating ready retrievability of the MPC after the tip-over. As an option, the radial channels may be fabricated from SA240-304 material. The difference in material properties, however, has a negligible effect on the end results.

Table 3.A.3: Key Input Data in Drop Analyses

Overpack weight	267,664 lb
Radial Concrete weight	163,673 lb
Length of the cask	231.25 inches
Diameter of the bottom plate	132.50 inches
Inside diameter of the cask shell	72.50 inches
Outside diameter of the cask shells	132.50 inches
MPC weight (including fuel)	88,857 lb
MPC height	190.5 inches
MPC diameter	68.375 inches
MPC bottom plate thickness	2.5 inches
MPC top plate thickness	9.5 inches

Table 3.A.4: Filtered Results for Drop and Tip-Over Scenarios for HI-STORM 100<sup>†</sup>

Drop Event	Max. Displacement (inch)		Impact Velocity (in/sec)	Max. Deceleration <sup>††</sup> at the Top of the (g's) Basket		Duration of Deceleration Pulse (msec)	
	Set A	Set B		Set A	Set B	Set A	Set B
End Drop for 11 inches	0.65	0.81	92.2	43.98	41.53	3.3	3.0
Non-Mechanistic Tip-over	4.25	5.61	304.03	42.85	39.91	2.3	2.0

<sup>†</sup> The passband frequency of the Butterworth filter is 350 Hz.

<sup>††</sup> The distance of the top of the fuel basket is 206" from the pivot point. The distance of the top of the cask is 231.25" from the pivot point. Therefore, all displacements, velocities, and accelerations at the top of the fuel basket are 89.08% of those at the cask top (206"/231.25").



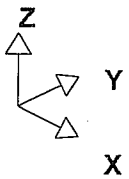
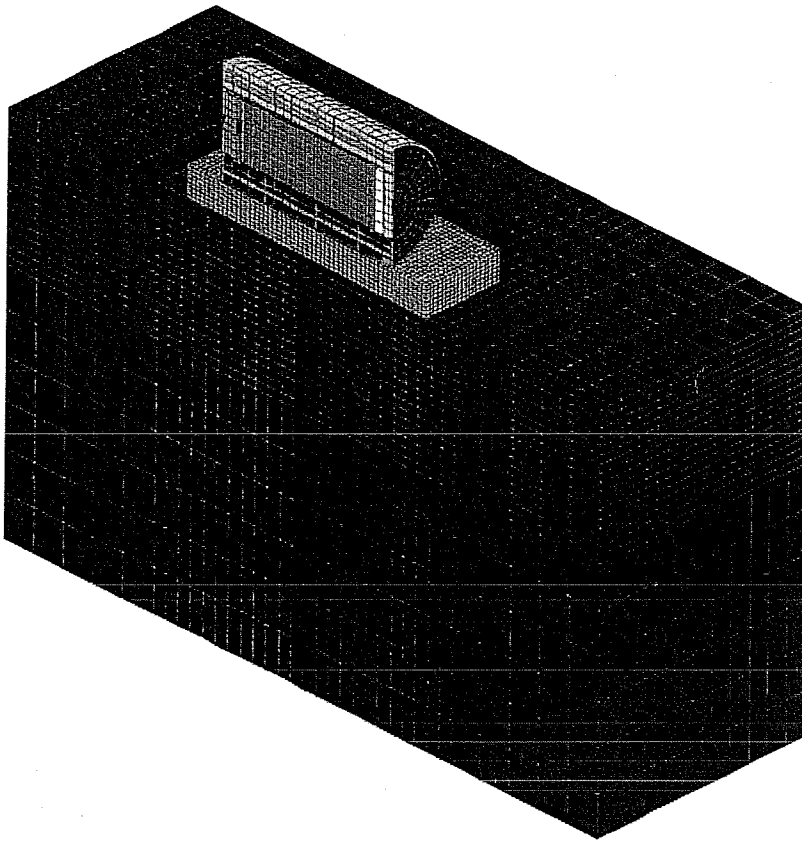


Fig 3.A.1 Tipover Finite-Element Model (3-D View)

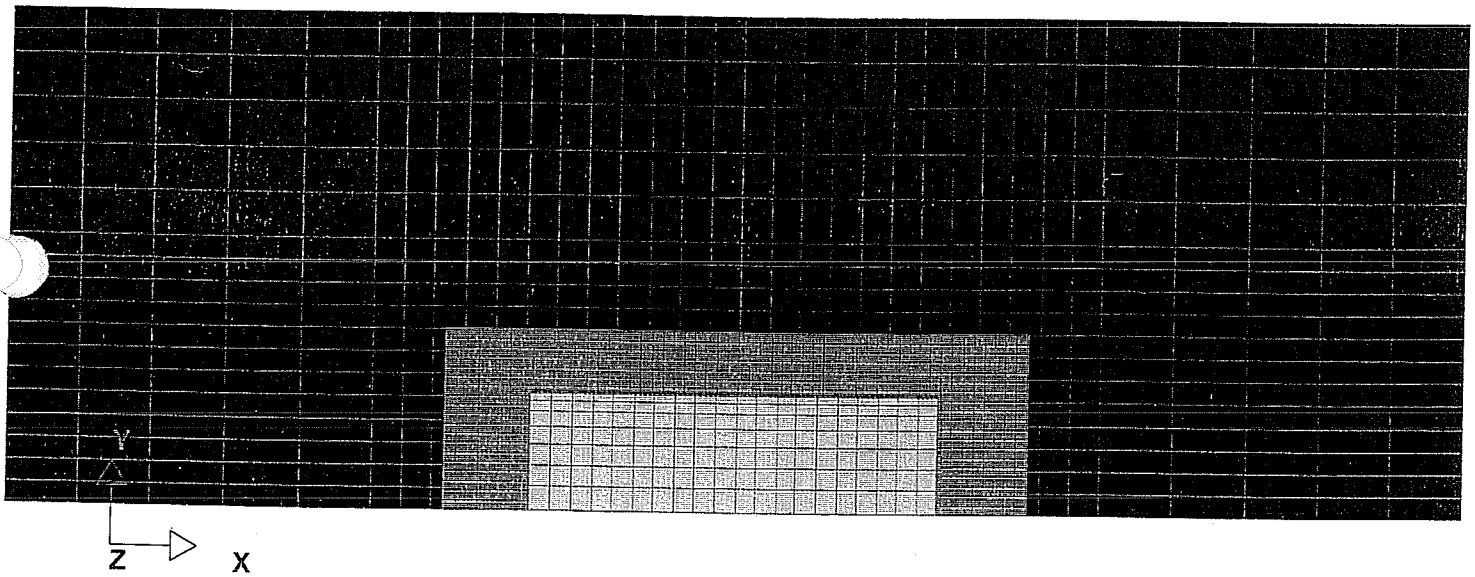


Fig 3.A.2 Tipover Finite-Element Model (Plan)

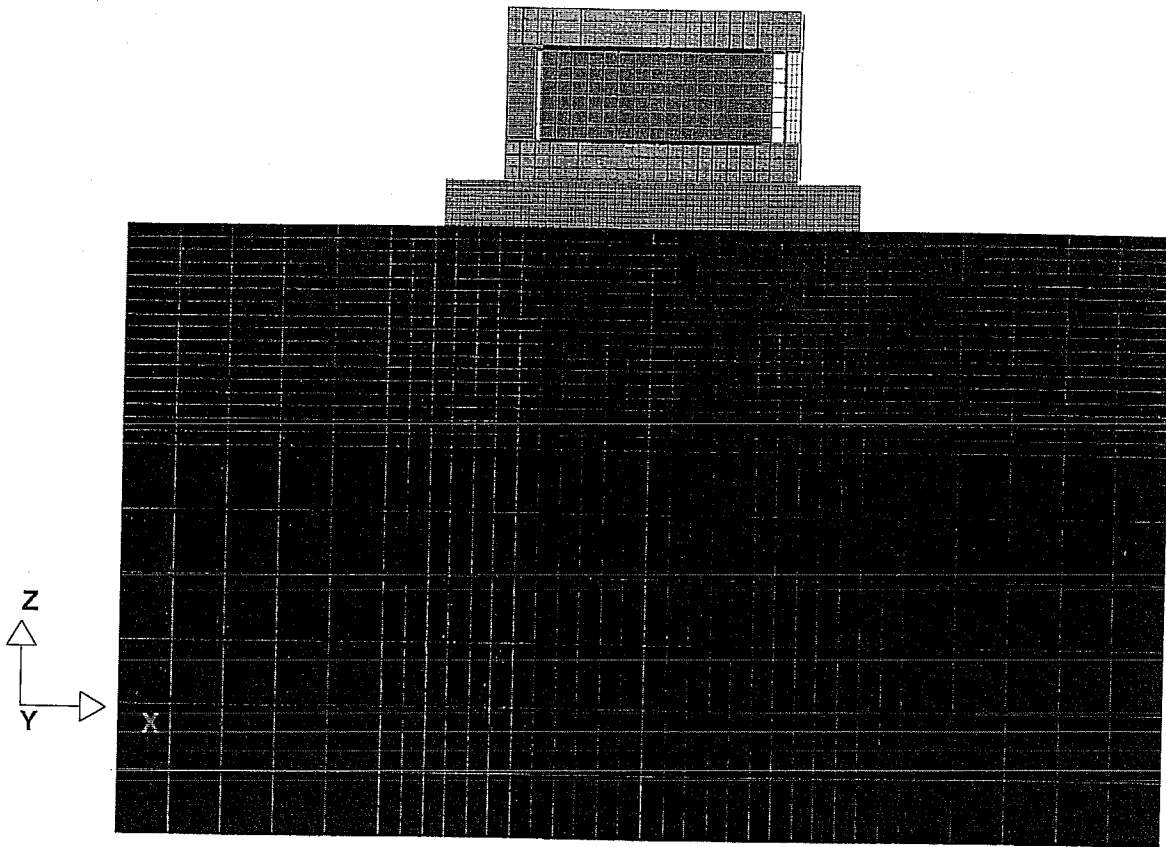


Fig 3.A.3 Tipover Finite-Element Model (XZView)

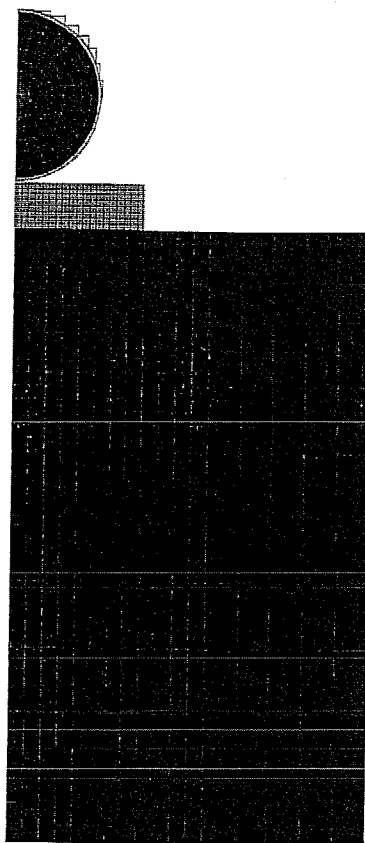
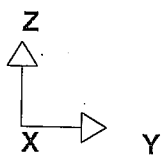


Fig 3.A.4 Tipover Finite-Element Model (YZ View)

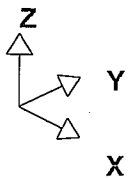
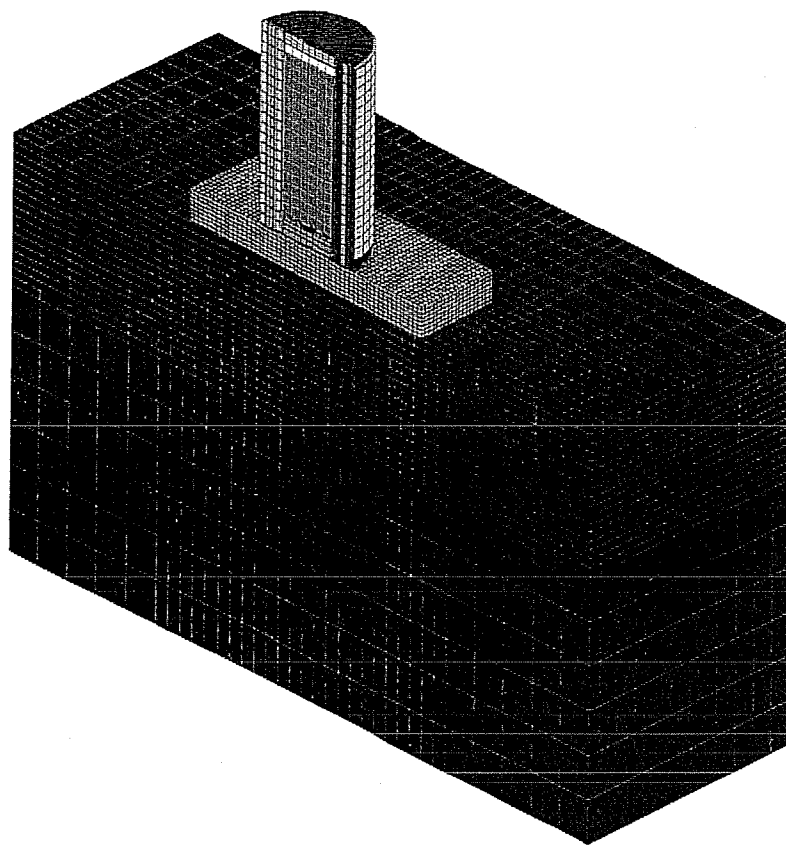


Fig 3.A.5 End-Drop Finite-Element Model (3-D View)

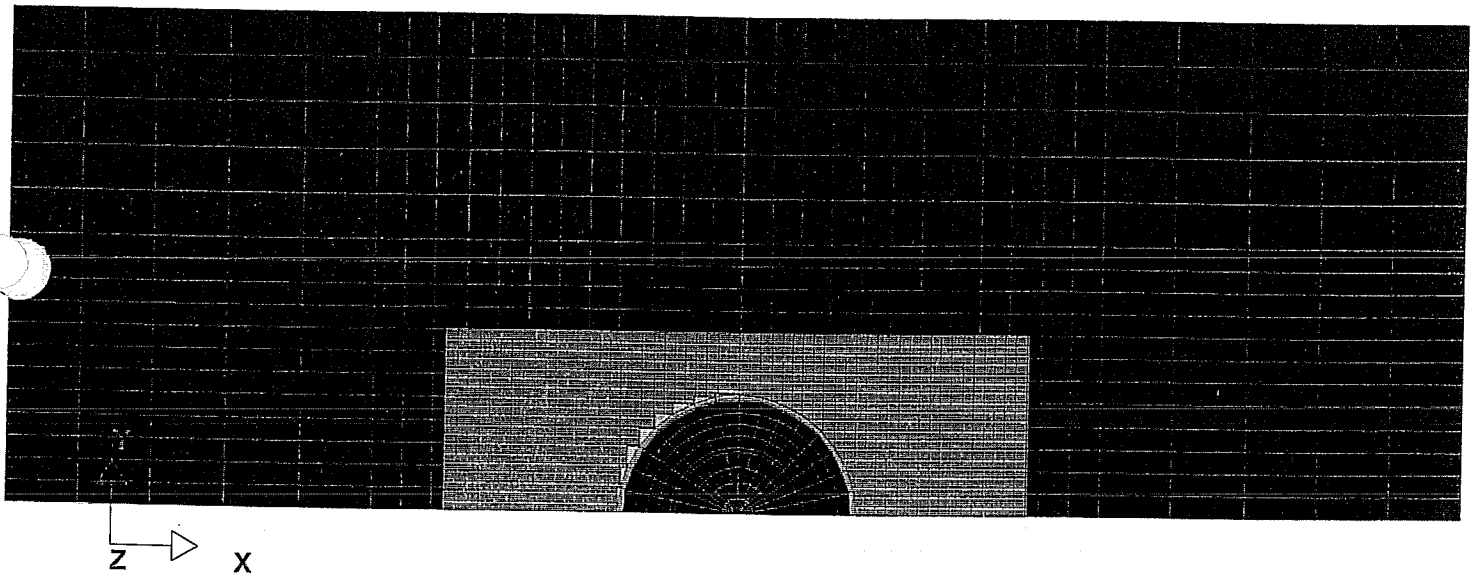


Fig 3.A.6 End-Drop Finite-Element Model (Plan)

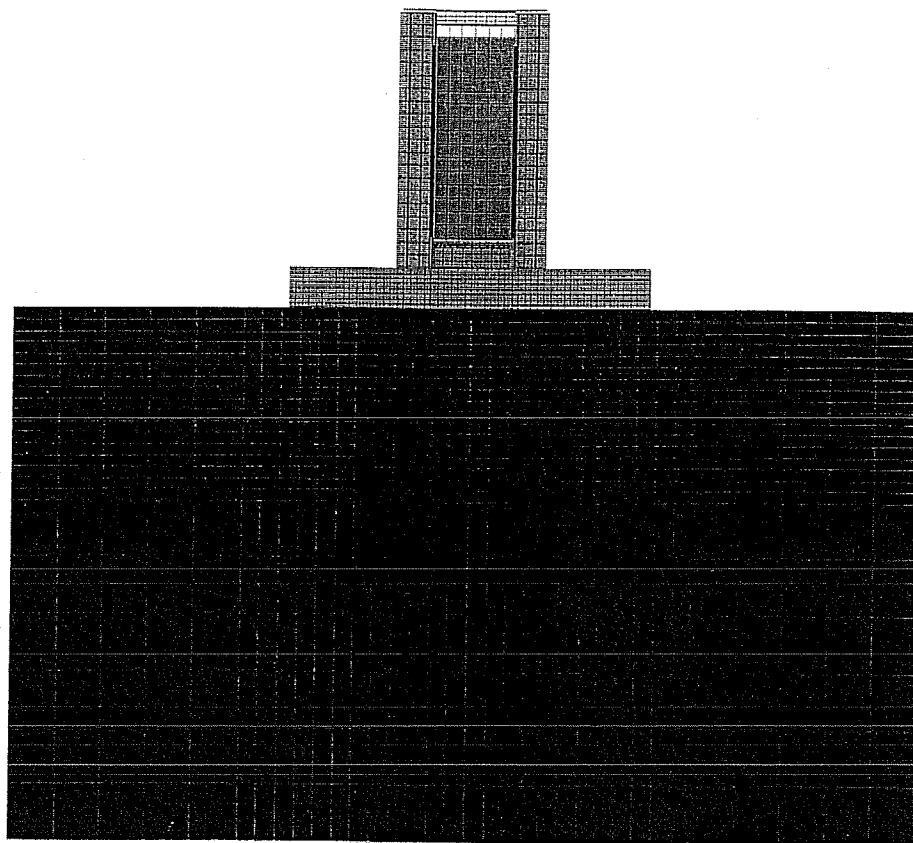
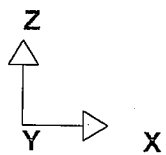


Fig 3.A.7 End-Drop Finite-Element Model (XZView)

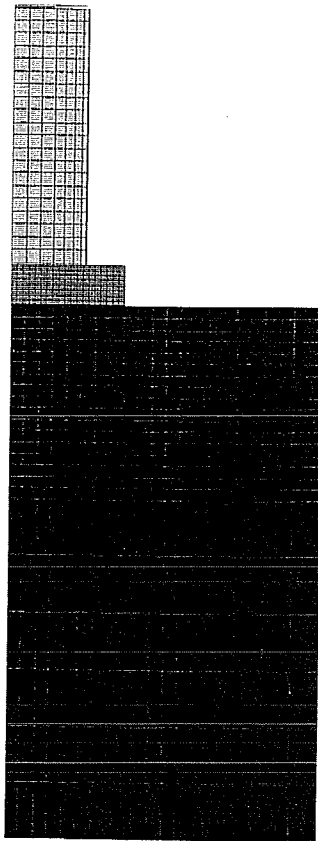
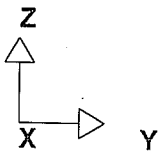


Fig 3.A.8 End-Drop Finite-Element Model (YZ View)



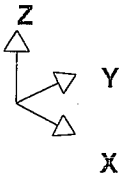
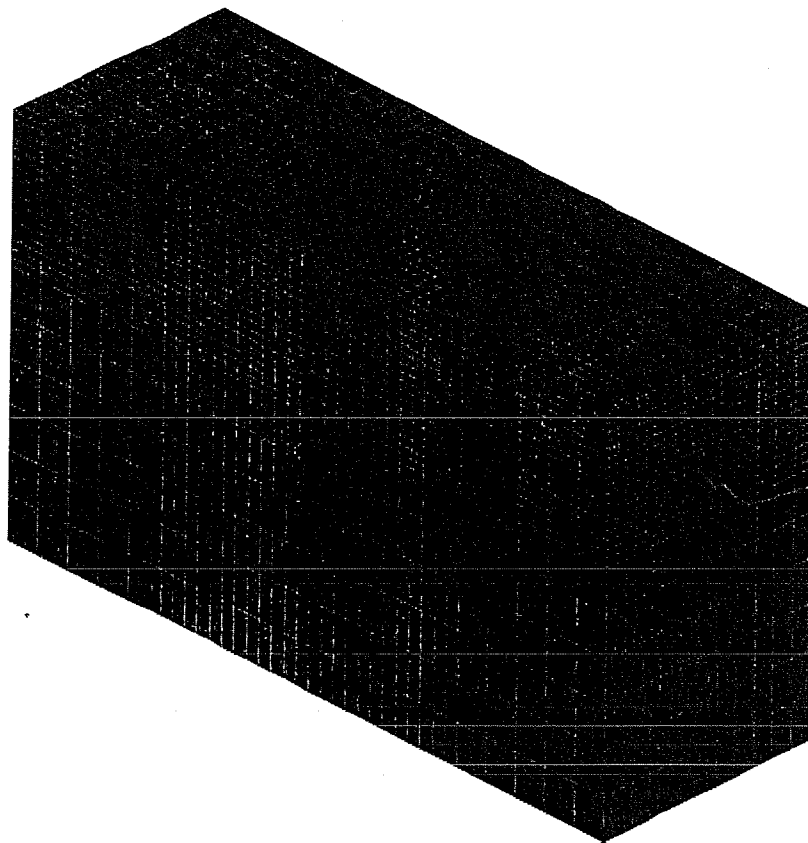
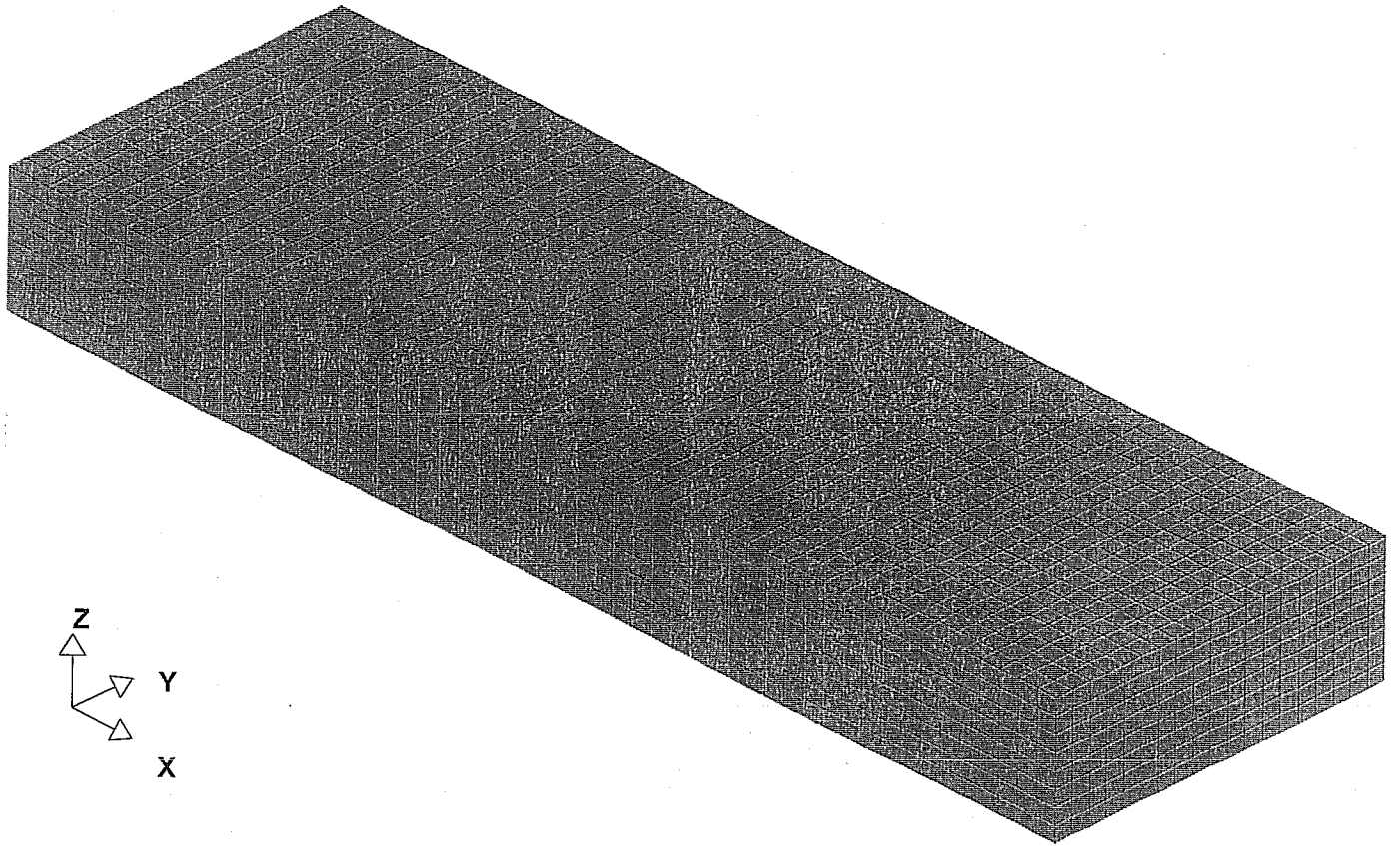


Fig 3.A.9 Soil Finite-Element Model (3-D View)



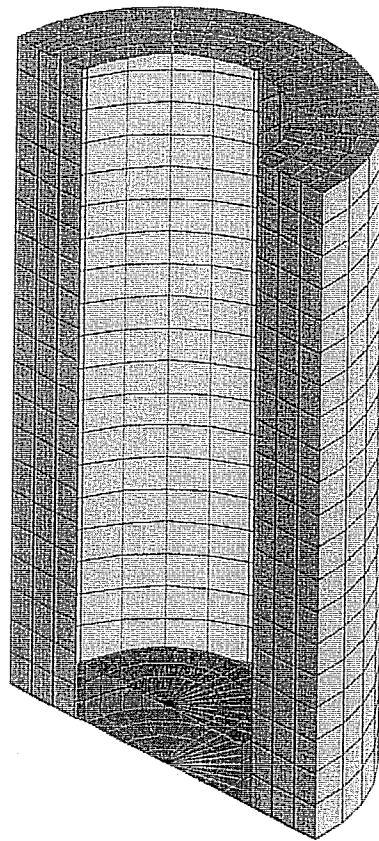
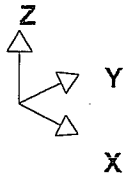


Fig 3.A.11 Overpack Steel Structure Finite-Element Model (3-D View)

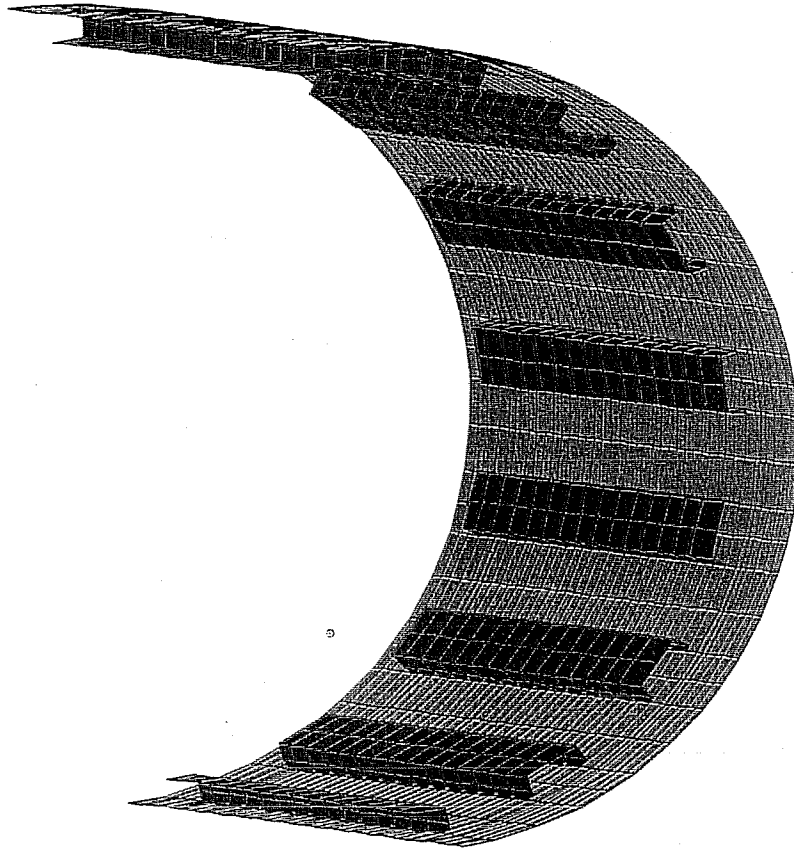
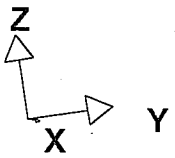


Fig. 3.A.12 Inner Shell and Channels Finite-Element Model (3-D View)

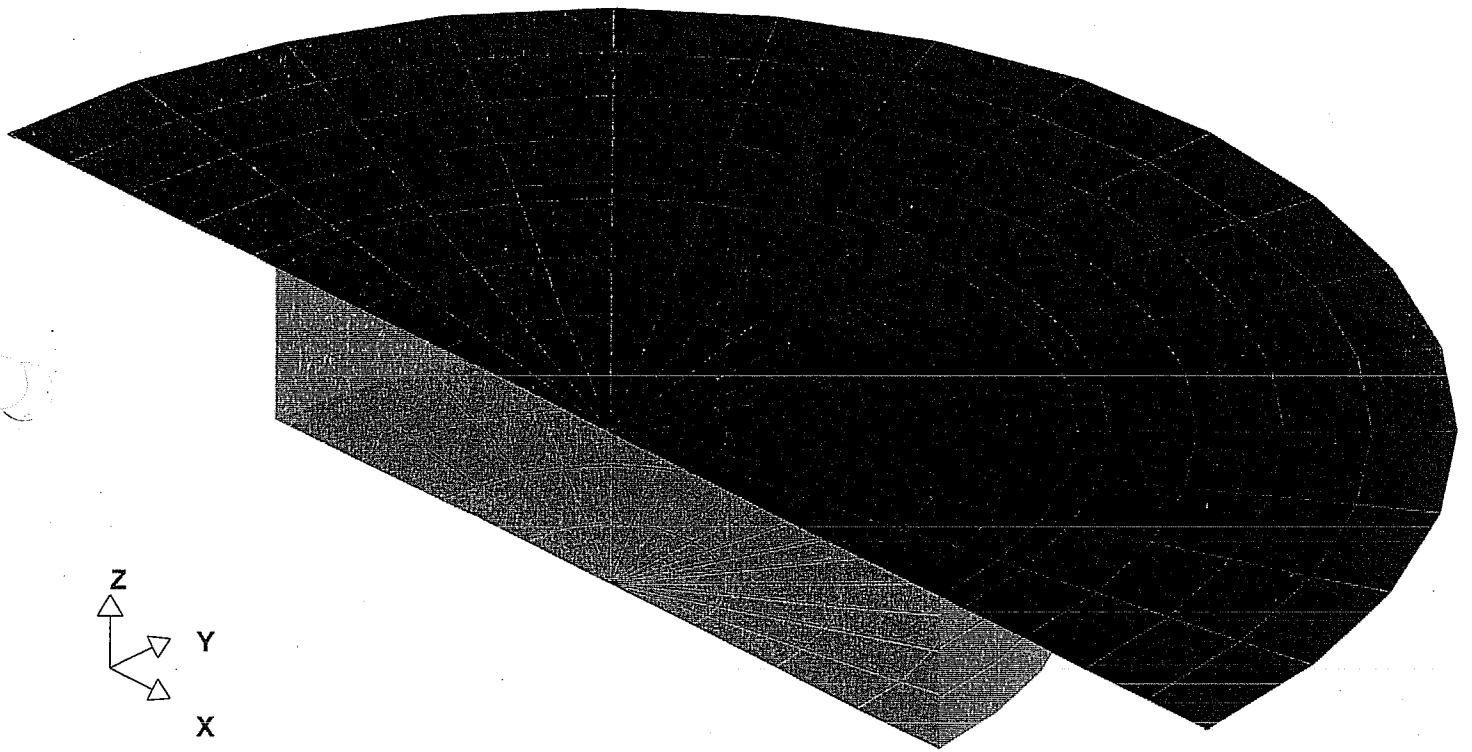


Fig 3.A.13 Lid Steel Finite-Element Model (3-D View)

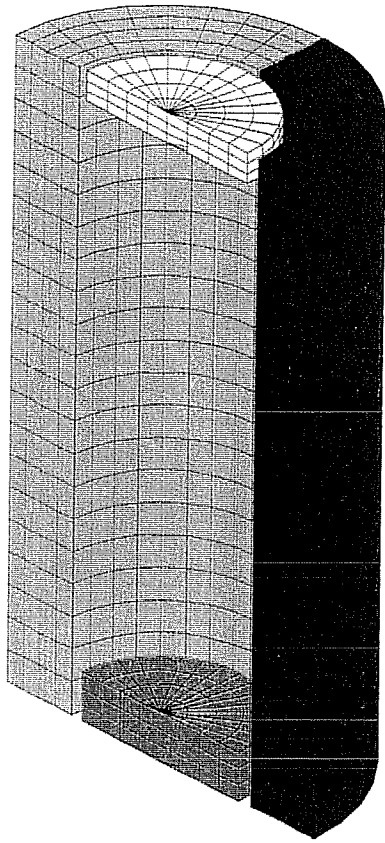
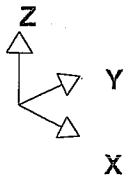
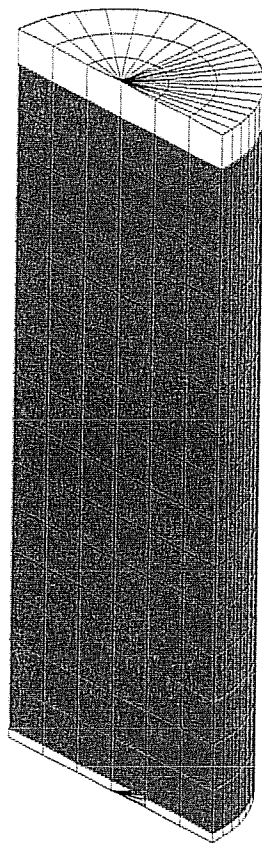
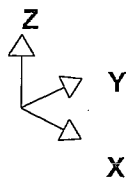


Fig 3.A.14 Overpack Concrete Components Finite-Element Model (3-D View)



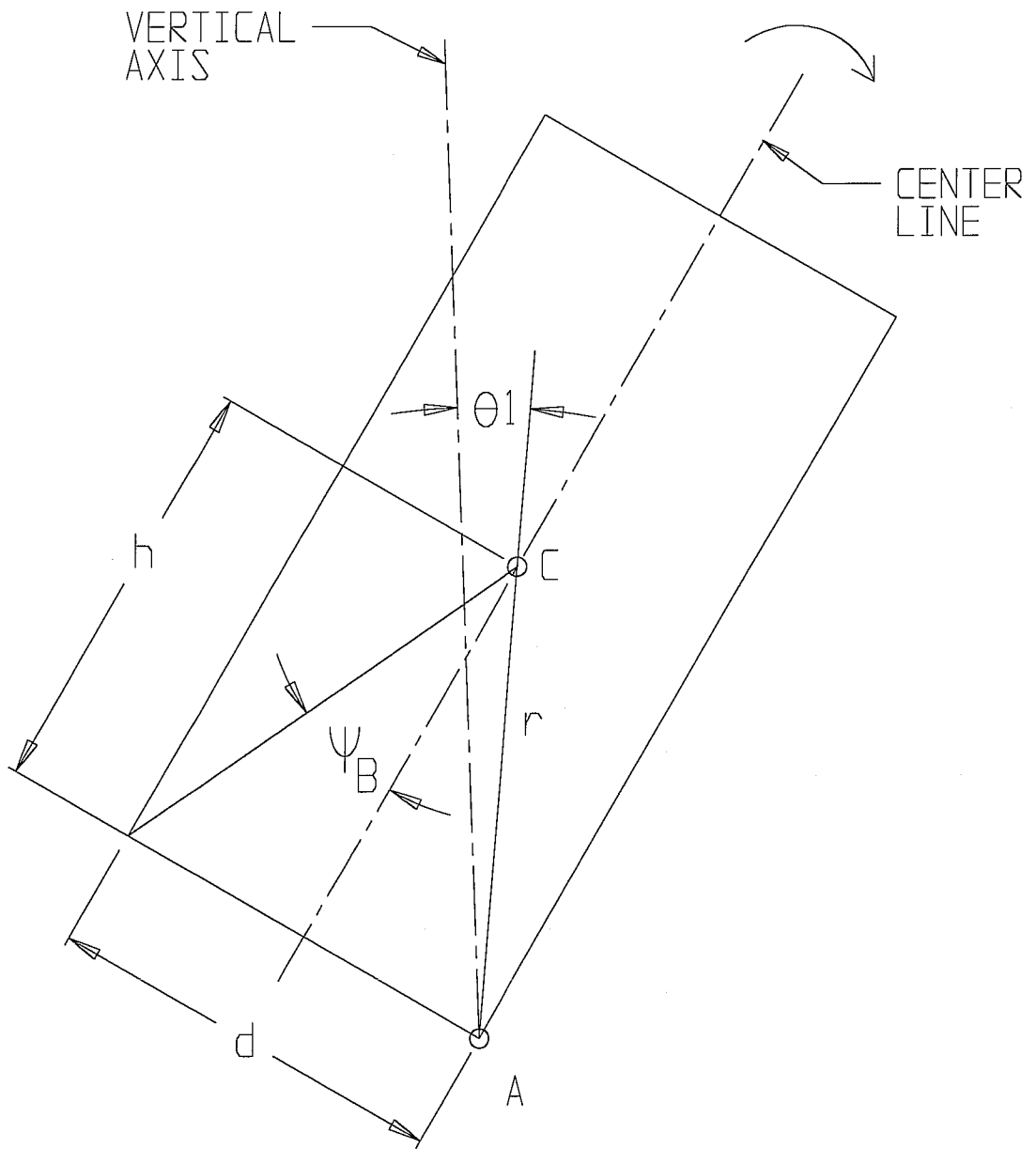


FIGURE 3.A.16; PIVOT POINT DURING TIP-OVER CONDITION



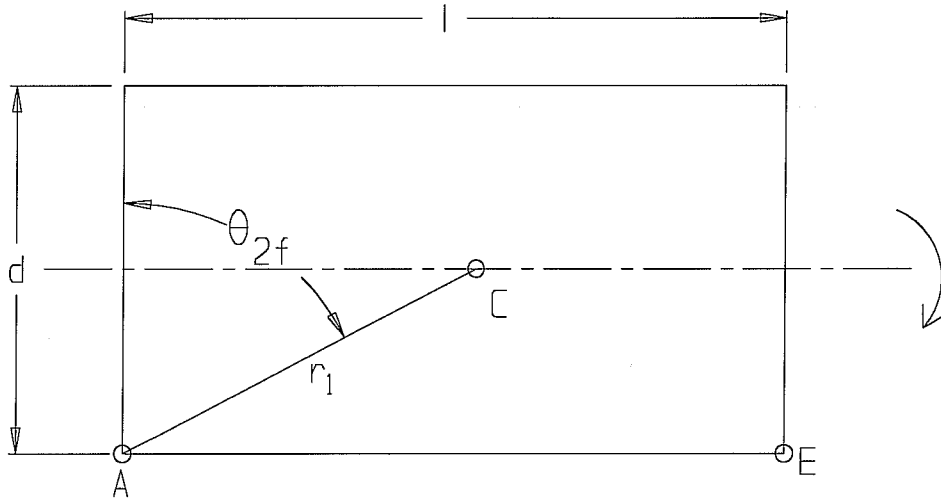


FIGURE 3.A.17; TIP-OVER EVENT OVERPACK SLAMS AGAINST THE FOUNDATION DEVELOPING A RESISTIVE FORCE

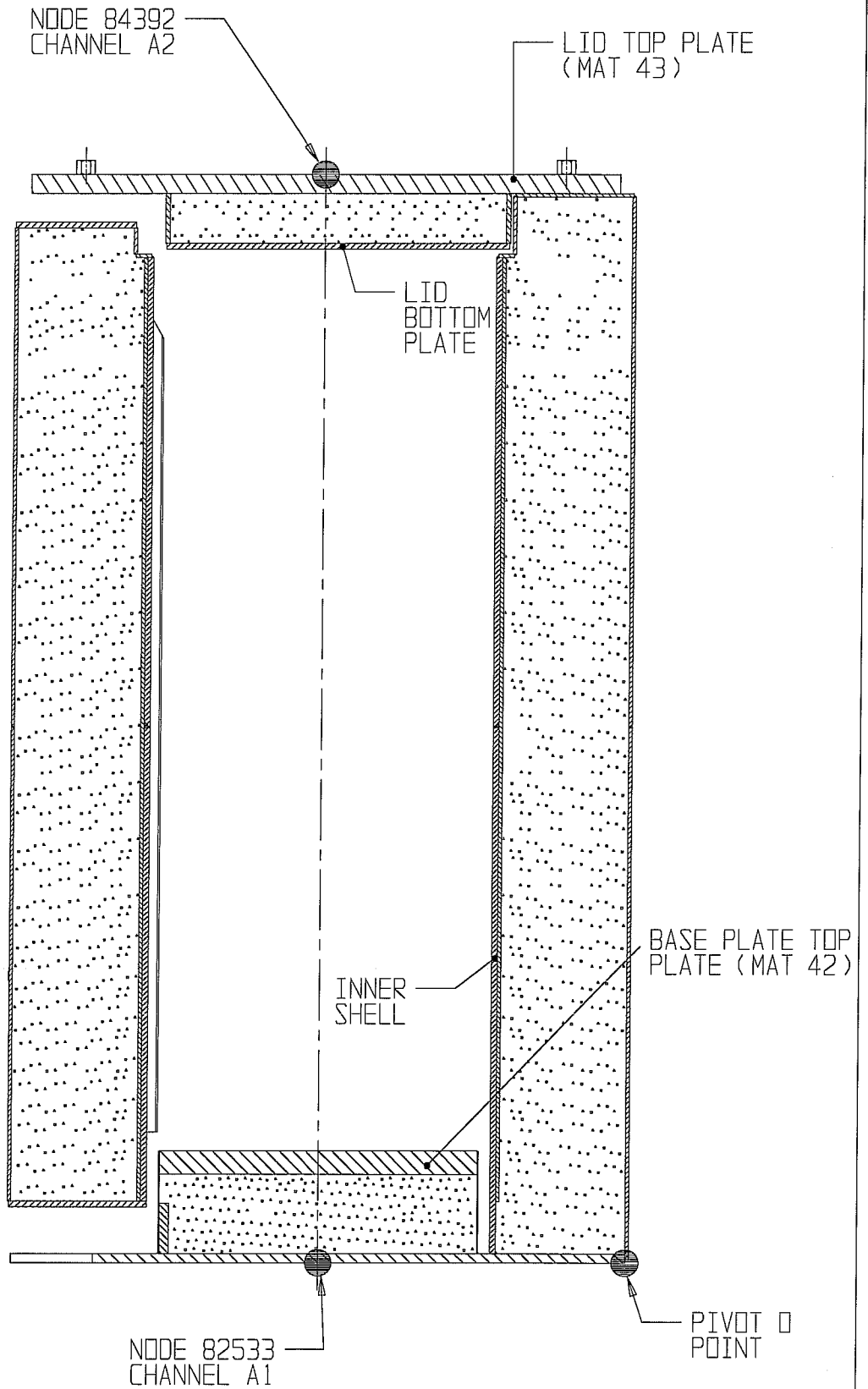


FIGURE 3.A.18; MEASUREMENT POINTS AND CORRESPONDING FINITE-ELEMENT MODEL NODES

## CHAPTER 4<sup>†</sup> 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. ~~Safe thermal performance during on-site loading, unloading and transfer operations utilizing the HI-TRAC transfer cask is also demonstrated.~~ The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under the most severe design basis ambient conditions. Effects 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 adopts guidelines presented in NUREG-1536 [4.4.10] and ISG-11 [4.1.4] guidelines to demonstrate safe storage of Commercial Spent Fuel (CSF)\*. ~~include eight specific acceptance criteria that should be fulfilled by the cask thermal design. These eight criteria are summarized here as follows~~ guidelines are stated below:*

1. The fuel cladding temperature *for long-term storage and short-term operations shall be limited to 400 °C (752 °F).* ~~at the beginning of dry cask storage should generally be below the anticipated damage threshold temperatures for normal conditions and a minimum of 20 years of cask storage.~~
2. The fuel cladding temperature should ~~generally~~ be maintained below 570<sup>0</sup>C (1058<sup>0</sup>F) for accident; *and off-normal event;* ~~and fuel transfer conditions.~~
3. The maximum internal pressure of the cask should remain within its design pressures for normal (1% rod rupture), off-normal (10% rod rupture), and accident (100% rod rupture) conditions.

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

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

4. The cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.
5. For fuel assemblies proposed for storage, the cask system should ensure a very low probability of cladding breach during long-term storage.
- ~~6. Fuel cladding damage resulting from creep cavitation should be limited to 15% of the original cladding cross-sectional area.~~
76. The ~~cask~~ HI-STORM sSystem should be passively cooled.
87. The thermal performance of the cask should be within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal, off-normal, and accident conditions.

As demonstrated in this chapter (see Subsections 4.4.6 and 4.5.6), the HI-STORM System is designed to comply with all eight of the criteria listed above. All thermal analyses to evaluate normal conditions of storage in a HI-STORM storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC transfer cask are described in Section 4.5. All analyses for off-normal conditions are described in Section 11.1. All analyses for accident conditions are described in Section 11.2. Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions. This FSAR chapter is in full compliance with NUREG-1536 requirements, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.3 and to ISG-11 requirements (no exceptions).

~~This revision to the HI-STORM Safety Analysis Report, the first since the HI-STORM 100 System was issued a Part 72 Certificate of Compliance, incorporates several features into the thermal analysis to respond to the changing needs of the U.S. nuclear power generation industry and revisions to NRC regulations. The most significant changes are:~~

- *The thermal analysis is revised to comply with recently issued NRC Staff Guidance (“Cladding Considerations for the Transportation and Storage of Spent Fuel,” ISG-11, Rev. 3).*
- *The Aluminum Heat Conduction Elements (ACHEs), optional under Amendment 1 of CoC 1014, are removed from the design. Removing the ACHEs from the MPC eliminates the constriction of the downcomer flow and thus further enhances the thermal performance of the MPC.*

~~Post-core decay time (PCDT) limitations on high burnup fuel (burnup > 45,000 MWD/MTU) have been computed. The allowable cladding temperatures for high burnup PWR and BWR fuel, required to establish PCDT limits, are computed using a methodology consistent with ISG-11.~~

- ~~□ Both uniform and regionalized storage are permitted, the latter being particularly valuable in mitigating the dose emitted by the MPC by restricting “cold and old” SNF in the locations surrounding the core region of the basket (where the “hot and new” fuel is stored).~~
- ~~□ The effect of convective heat transfer in the MPC, originally included in the analysis but subsequently neglected to enable the NRC to make a more considered assessment of gravity-driven convective heat transfer in honeycomb basket equipped MPCs, is now reintroduced.~~
- ~~□ In the absence of the credit for convective (thermosiphon) effect, the previous analysis relied on the conduction heat transfer through the clearance between the basket and the MPC enclosure vessel. The conduction heat flow path was provided by the Aluminum Heat Conduction Elements (AHCE). The AHCE hardware is retained in the MPC and credit for ACHE heat dissipation is eliminated in the thermal analyses to maintain a solid margin of conservatism in the computed results. In a similar spirit of conservatism, the heat transfer in narrow cavities (the Rayleigh effect), approved by the SFPO in the previous analysis, is neglected in this revision.~~

~~Aside from the above-mentioned changes, this revision of this chapter is essentially identical to its predecessor.~~

## 4.1 DISCUSSION

*As discussed in Chapter 2, this revision of the HI-STORM FSAR seeks to establish complete compliance with the provisions of ISG-11 [4.1.4]. To ensure explicit compliance, the new condition “short term operations,” corresponding to fuel loading activities, is defined in Chapter 2.*

*In Revision 1 of this FSAR, fuel loading, which includes MPC cavity drying, MPC lid welding, helium pressurization, and MPC transfer operations, was treated as part of the “off-normal” condition. It is not treated as a distinct fuel thermal state. Specifically, the maximum fuel cladding temperature for the fuel loading condition now formally referred to as “short term operations” is set equal to the PCT limit for normal storage conditions for all high-burnup CSF (see Section 4.3). Potential thermally challenging states for the spent fuel arise if the fuel drying process utilizes pressure reduction (i.e., vacuum drying) or when the loaded MPC is inside the HI-TRAC transfer cask. In the latter state, the rate of heat rejection from the MPC is somewhat less compared to the normal storage condition when the MPC is inside the ventilated overpack. Because the HI-TRAC transfer cask handling subsequent to helium pressurization of the MPC typically involves keeping the equipment vertical, the thermosiphon action inside the MPC is fully operational during these activities. As a result, the increase in the fuel cladding temperature in the HI-TRAC compared to the HI-STORM storage condition is fairly modest. The increase is more significant in the case where the HI-TRAC transfer cask, for reasons such as vertical height restrictions or seismic constraints at a plant, must be handled in the horizontal orientation. When the HI-TRAC is horizontal, the cessation of the thermosiphon action results in an additional rise in the fuel cladding temperature. Therefore, the short term evolutions that may be thermally limiting are analyzed as listed below:*

- i. Vacuum Drying*
- ii. Loaded MPC in HI-TRAC in the vertical orientation*
- iii. Loaded MPC in HI-TRAC in the horizontal orientation*

*The threshold MPC heat generation rate at which the HI-STORM peak cladding temperature reaches a steady state equilibrium value approaching the normal storage peak clad temperature limit is computed in this chapter. Likewise, the MPC heat generation rates that produce the steady state equilibrium temperature approaching the normal storage peak clad temperature limit for the MPC-in-HI-TRAC condition in both vertical and horizontal configurations are computed in this chapter. These computed heat generation rates directly bear upon the compliance of the system with ISG-11 [4.1.4] and are, accordingly, adopted in the system Technical Specifications for high burnup fuel (HBF).*

*A sectional cutaway view of the HI-STORM dry storage system has been presented earlier (see Figure 1.2.1). The system consists of a sealed MPC situated inside a vertical ventilated storage overpack. Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top, respectively, of the cylindrical overpack. The SNF assemblies reside inside the MPC, which is sealed with a welded lid to form the confinement boundary. The MPC contains an all-alloy honeycomb basket structure with square-shaped compartments of appropriate dimensions to*

allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. Each box panel, with the exception of exterior panels on the MPC-68 and MPC-32, is equipped with a ~~Boral~~ (thermal neutron absorber) panel sandwiched between an ~~Alloy X~~ steel sheathing plate and the box panel, along the entire length of the active fuel region. The MPC is backfilled with helium up to the design-basis initial fill level (Table 1.2.2). This provides a stable, inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM System to the environment by passive heat transport mechanisms only.

The helium backfill gas is an integral part of the MPC thermal design. The helium fills all the spaces between solid components and provides an improved conduction medium (compared to air) for dissipating decay heat in the MPC. ~~Additionally, helium in the spaces between the fuel basket and the MPC shell is heated differentially and, therefore, subject to the “Rayleigh” effect which is discussed in detail later. For added conservatism, the increase in the heat transfer rate due to the Rayleigh effect contribution is neglected in this revision of the FSAR.~~ To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC confinement boundary is designed and fabricated to comply with the provisions of the ASME B&PV Code Section III, Subsection NB (to the maximum extent practical), as an all-seal-welded pressure vessel with redundant closures. It is demonstrated in Section 11.1.3 that the failure of one field-welded pressure boundary seal will not result in a breach of the pressure boundary. The helium gas is therefore retained and undiluted, and may be credited in the thermal analyses.

An important thermal design criterion imposed on the HI-STORM System is to limit the maximum fuel cladding temperature to within design basis limits (Table 4.3.17) for long-term storage of design basis SNF assemblies. An equally important design criterion is to minimize temperature gradients in the MPC so as to minimize thermal stresses. In order to meet these design objectives, the MPC baskets are designed to possess certain distinctive characteristics, which are summarized in the following.

The MPC design minimizes resistance to heat transfer within the basket and basket periphery regions. This is ensured by an uninterrupted panel-to-panel connectivity realized in the all-welded honeycomb basket structure. The MPC design incorporates top and bottom plenums with interconnected downcomer paths. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket, and by elongated semicircular holes in each basket cell wall. The bottom plenum is formed by large elongated semicircular holes at the base of all cell walls. The MPC basket is designed to eliminate structural discontinuities (i.e., gaps) which introduce large thermal resistances to heat flow. Consequently, temperature gradients are minimized in the design, which results in lower thermal stresses within the basket. Low thermal stresses are also ensured by an MPC design that permits unrestrained axial and radial growth of the basket. The possibility of stresses due to restraint on basket periphery thermal growth is eliminated by providing adequate basket-to-canister shell gaps to allow for basket thermal growth during heat-up to design basis temperatures.

It is heuristically apparent from the geometry of the MPC that the basket metal, the fuel assemblies, and the contained helium mass will be at their peak temperatures at or near the longitudinal axis of the MPC. The temperatures will attenuate with increasing radial distance from this axis, reaching their lowest values at the outer surface of the MPC shell. Conduction along the metal walls and radiant heat exchange from the fuel assemblies to the MPC metal mass would therefore result in substantial differences in the bulk temperatures of helium columns in different fuel storage cells. Since two fluid columns at different temperatures in communicative contact cannot remain in static equilibrium, the non-isotropic temperature field in the MPC internal space due to conduction and radiation heat transfer mechanisms guarantee the incipience of the third mode of heat transfer: natural convection.

The preceding paragraph introduced the internal helium thermosiphon feature engineered into the MPC design. It is recognized that the backfill helium pressure, in combination with low pressure drop circulation passages in the MPC design, induces a thermosiphon upflow through the multicellular basket structure to aid in removing the decay heat from the stored fuel assemblies. The decay heat absorbed by the helium during upflow through the basket is rejected to the MPC shell during the subsequent downflow of helium in the peripheral downcomers. This helium thermosiphon heat extraction process significantly reduces the burden on the MPC metal basket structure for heat transport by conduction, thereby minimizing internal basket temperature gradients and resulting thermal stresses.

The helium columns traverse the vertical storage cavity spaces, redistributing heat within the MPC. Elongated holes in the bottom of the cell walls, liberal flow space and elongated holes at the top, and wide-open downcomers along the outer periphery of the basket ensure a smooth helium flow regime. The most conspicuous beneficial effect of the helium thermosiphon circulation, as discussed above, is the mitigation of internal thermal stresses in the MPC. Another beneficial effect is reduction of the peak fuel cladding temperatures of the fuel assemblies located in the interior of the basket. ~~In the original HI-STORM licensing analyses, no credit for the thermosiphon action was taken. To partially compensate for the reduction in the computed heat rejection capability due to the complete neglect of the global thermosiphon action within the MPC, heat conduction elements made of aluminum were interposed in the large peripheral spaces between the MPC shell and the fuel basket. These heat conduction elements, shown in the MPC Drawings in Section 1.5, are engineered such that they can be installed in the peripheral spaces to create a nonstructural thermal connection between the basket and the MPC shell. In their installed condition, the heat conduction elements will contact the MPC shell and the basket walls. MPC manufacturing procedures have been established to ensure that the thermal design objectives for the conduction elements set forth in this document are realized in the actual hardware. The presence of heat conduction elements in the canister design has been conservatively neglected in the thermal models of the HI-STORM 100 System in this revision of the Safety Analysis Report.~~

Four distinct MPC basket geometries are evaluated for thermal performance in the HI-STORM System. For intact PWR fuel storage, the MPC-24, MPC-24E, and MPC-32 designs are available.



Four locations are designated for storing damaged PWR fuel in the MPC-24E design. A 68-cell MPC design (MPC-68, MPC-68F, and MPC-68FF) is available for storing BWR fuel (intact or damaged (including fuel debris)). All of the four basic MPC geometries (MPC-32, MPC-24, MPC-24E and MPC-68) are described in Chapter 1 wherein their ~~design~~ licensing drawings can also be found.

The design maximum decay heat loads for storage of intact zircaloy clad fuel in the four MPCs are listed in Tables 4.4.20, 4.4.21, 4.4.28, and 4.4.29. Storage of intact stainless steel *clad fuel* is ~~permitted for a low decay heat limit set forth in Chapter 2 (Tables 2.1.17 through 2.1.21) evaluated in Subsection 4.3.2.—~~ Storage of zircaloy clad fuel with stainless steel clad fuel in an MPC is permitted. In this scenario, the zircaloy clad fuel is ~~conservatively stipulated to~~ *must* meet the lower decay heat limits for stainless steel clad fuel. Storage of damaged, zircaloy clad fuel is evaluated in Subsection 4.4.1.1.4.— The axial heat distribution in each fuel assembly is assumed to follow the burnup profiles set forth by Table 2.1.11.

Thermal analysis of the HI-STORM System is based on including all three fundamental modes of heat transfer, namely conduction, natural convection and radiation. Different combinations of these modes are active in different parts of the system. These modes are properly identified and conservatively analyzed within each part of the MPC, the HI-STORM storage overpack and the HI-TRAC transfer cask, to enable bounding calculations of the temperature distribution within the HI-STORM System to be performed. In addition to storage within the HI-STORM overpack, loaded MPCs will also be located for short durations inside the transfer cask (HI-TRAC) designed for moving MPCs into and out of HI-STORM storage modules.

Heat is dissipated from the outer surface of the *HI-STORM* storage overpack and HI-TRAC *transfer cask* to the environment by buoyancy induced airflow (natural convection) and thermal radiation. Heat transport through the cylindrical wall of the storage overpack and HI-TRAC is solely by conduction. While stored in a HI-STORM overpack, heat is rejected from the surface of the MPC via the parallel action of thermal radiation to the inner shell of the overpack and convection to a buoyancy driven airflow in the annular space between the outer surface of the MPC and the inner shell of the overpack. This situation is similar to the familiar case of natural draft flow in furnace stacks. When placed into a HI-TRAC cask for transfer operations, heat is rejected from the surface of the MPC to the inner shell of the HI-TRAC by conduction and thermal radiation.

Within the MPC, heat is transferred between metal surfaces (e.g., between neighboring fuel rod surfaces) via a combination of conduction through a gaseous medium (helium) and thermal radiation. Heat is transferred between the fuel basket and the MPC shell by thermal radiation and conduction. ~~The heat transfer between the fuel basket external surface and the MPC shell inner surface is further influenced by the “Rayleigh” effect. The heat transfer augmentation effect of this mechanism, as discussed earlier, is conservatively neglected.~~

As discussed later in this chapter, an array of conservative assumptions bias the results of the

thermal analysis towards much reduced computed margins than would be obtained by a rigorous analysis of the problem. ~~In particular, the thermal model employed in determining the MPC temperatures is consistent with the model presented in Rev. 9 of the HI-STAR FSAR submittal (Docket No. 72-1008).~~

~~As discussed in Chapter 2, the HI-STORM MPCs are identical to those utilized in the NRC-accepted HI-STAR System (Docket 71-1008 for storage). As such, many of the analysis methods utilized herein for performing thermal evaluations of the HI-STORM MPCs are identical to those already accepted for the HI-STAR System. Specifically, the analysis methods for evaluation of the following items are identical to those for the HI-STAR System:~~

- ~~i. fuel assembly effective thermal conductivity~~
- ~~ii. MPC fuel basket effective thermal conductivity~~
- ~~iii. MPC fuel basket peripheral region effective thermal conductivity~~
- ~~iv. aluminum heat conduction elements effective thermal conductivity~~
- ~~v. MPC internal cavity free volume~~
- ~~vi. MPC contents effective heat capacity and density~~
- ~~vii. bounding fuel rod internal pressures and hoop stresses~~

~~In addition, thermal properties for all materials common to both the HI-STORM and HI-STAR systems are identical, including stainless and carbon steels, zircaloy, UO<sub>2</sub>, aluminum alloy 1100, Boral, Holtite A, helium, air and paint.~~

The complete thermal analysis is performed using the industry standard ANSYS finite element modeling package [4.1.1] and the finite volume Computational Fluid Dynamics (CFD) code FLUENT [4.1.2]. ANSYS has been previously used and accepted by the NRC on numerous dockets [4.4.10, 4.5.5.a]. The FLUENT CFD program is independently benchmarked and validated with a wide class of theoretical and experimental studies reported in the technical journals. Additionally, Holtec has confirmed the code's capability to reliably predict temperature fields in dry storage applications *in a benchmark report [4.1.5]* using independent full-scale test data from a loaded cask [4.1.3]. ~~A series of Holtec topical reports, culminating in "Topical report on the HI-STAR/HI-STORM thermal model and its benchmarking with full size cask test data", Holtec Report HI-992252, Rev. 1, document the comparison of the Holtec thermal model against the full size cask test data [4.1.3]. In this benchmarking report reference [4.1.3], the Holtec thermal model is shown to overpredict the measured fuel cladding temperature by a modest amount for every test set. In early 2000, PNL evaluated the thermal performance of HI-STORM 100 at discrete ambient temperatures using the COBRA-SFS Code. (Summary report communicated by T.E. Michener to J. Guttman (NRC staff) dated May 31, 2000 titled "TEMPEST Analysis of the Utah ISFSI Private Fuel Storage Facility and COBRA-SFS Analysis of the Holtec HI-STORM 100 Storage System"). The above-mentioned topical benchmarking report has been updated to include a comparison of the Holtec thermal model results with the PNL solution. Once again, *the comparison shows that* Holtec thermal model *is continues to be* uniformly conservative, ~~albeit by small margins.~~ The benchmarking~~

of the Holtec thermal model [4.1.5] against the EPRI test data [4.1.3] and PNL COBRA-SFS study |  
validate the suitability of the thermal model employed to evaluate the thermal performance of the  
HI-STORM 100 System in this document.

## 4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Materials present in the MPCs include stainless steels (Alloy X), Boral-neutron absorber (*Boral* or *METAMIC*), aluminum Alloy 1100 heat conduction elements, and helium. Materials present in the HI-STORM storage overpack include carbon steels and concrete. Materials present in the HI-TRAC transfer cask include carbon steels, lead, Holtite-A neutron shield, and demineralized water<sup>†</sup>. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Individual thermal conductivities of the alloys that comprise the Alloy X materials and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2; and 4.2.3 and 4.2.9 provide numerical thermal conductivity data of materials at several representative temperatures. Thermal conductivity data for *constituents of Boral-components* (i.e., B<sub>4</sub>C core and aluminum cladding) is provided in Table 4.2.8. *Boral is a compressed neutron absorbing core with a thin layer of aluminum on both sides. Because of its sandwich construction, its conduction properties are directionally dependent (i.e., non-isotropic). In contract to Boral, METAMIC is a homogeneous neutron absorbing material with a thermal conductivity that is higher than the Boral neutron absorbing B4C core (Figure 4.2.3) but lower than Boral's aluminum cladding. The equivalent conductivity of a Boral panel, defined as the Square Root of the Mean Sum of Squares (SRMSS) conductivity in two principal directions (through thickness and width) is closely matched by METAMIC<sup>‡</sup>. Therefore, the two materials are considered equivalent in their heat transfer performance. The temperature dependence of the thermal conductivities of helium and air is shown in Figure 4.2.1.*

For the HI-STORM overpack, the thermal conductivity of concrete and the emissivity/absorptivity of painted surfaces are particularly important. Recognizing the considerable variations in reported values for these properties, we have selected values that are conservative with respect to both authoritative references and values used in analyses on previously licensed cask docket. Specific discussions of the conservatism of the selected values are included in the following paragraphs.

As specified in Table 4.2.1, the concrete thermal conductivity is taken from Marks' Standard Handbook for Mechanical Engineers, which is conservative compared to a variety of recognized concrete codes and references. Neville, in his book "Properties of Concrete" (4<sup>th</sup> Edition, 1996), gives concrete conductivity values as high as 2.1 Btu/(hr×ft×°F). For concrete with siliceous aggregates, the type to be used in HI-STORM overpacks, Neville reports conductivities of at least 1.2 Btu/(hr×ft×°F). Data from Loudon and Stacey, extracted from Neville, reports conductivities of 0.980 to 1.310 Btu/(hr×ft×°F) for normal weight concrete protected from the weather. ACI-207.1R provides thermal conductivity values for seventeen structures (mostly dams) at temperatures from

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<sup>†</sup> Water from a primary source (e.g., lake or river) from which ionic impurities and precipitates have been removed.

<sup>‡</sup> For example, at 482 °F, the through-thickness and width direction conductivities of Boral (B<sub>4</sub>C thickness fraction = 0.82) are computed as 52.9 and 58.2 Btu/hr-ft-°F respectively. The SRMSS conductivity =  $[(52.9^2 + 58.2^2)/2]^{0.5}$  is 55.61 Btu/hr-ft-°F compared to a lowerbound METAMIC conductivity (Figure 4.2.3) of 55.68 Btu/hr-ft-°F (at 482 °F).

50-150°F. Every thermal conductivity value reported in ACI-207.1R is greater than the 1.05 Btu/(hr×ft×°F) value used in the HI-STORM thermal analyses.

Additionally, the NRC has previously approved analyses that use higher conductivity values than those applied in the HI-STORM thermal analysis. For example, thermal calculations for the NRC approved Vectra NUHOMS cask system (June 1996, Rev. 4A) used thermal conductivities as high as 1.17 Btu/(hr×ft×°F) at 100°F. Based on these considerations, the concrete thermal conductivity value stipulated for HI-STORM thermal analyses is considered to be conservative.

Holtite-A is a composite material consisting of approximately 37 wt% epoxy polymer, 1 wt% B<sub>4</sub>C and 62 wt% Aluminum trihydrate. Thermal conductivity of the polymeric component is low because polymers are generally characterized by a low conductivity (0.05 to 0.2 Btu/ft-hr-°F). Addition of fillers in substantial amounts raises the mixture conductivity up to a factor of ten. Thermal conductivity of epoxy filled resins with Alumina is reported in the technical literature† as approximately 0.5 Btu/ft-hr-°F and higher. In the HI-STORM FSAR, a conservatively postulated conductivity of 0.3 Btu/ft-hr-°F is used in the thermal models for the neutron shield region (in the HI-TRAC transfer cask). As the thermal inertia of the neutron shield is not credited in the analyses, the density and heat capacity properties are not reported herein.

Surface emissivity data for key materials of construction are provided in Table 4.2.4. The emissivity properties of painted external surfaces are generally excellent. Kern [4.2.5] reports an emissivity range of 0.8 to 0.98 for a wide variety of paints. In the HI-STORM thermal analysis, an emissivity of 0.85<sup>††</sup> is applied to painted surfaces. A conservative solar absorptivity coefficient of 1.0 is applied to all exposed overpack surfaces.

In Table 4.2.5, the heat capacity and density of the different MPC, overpack and CSF materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. The temperature dependence of the viscosities of helium and air are provided in Table 4.2.6 and plotted in Figure 4.2.2.

The heat transfer coefficient for exposed surfaces is calculated by accounting for both natural convection and thermal radiation heat transfer. The natural convection coefficient depends upon the product of Grashof (Gr) and Prandtl (Pr) numbers. Following the approach developed by Jakob and Hawkins [4.2.9], the product Gr×Pr is expressed as  $L^3 \Delta T Z$ , where L is height of the overpack,  $\Delta T$  is overpack surface temperature differential and Z is a parameter based on air properties, which are known functions of temperature, evaluated at the average film temperature. The temperature dependence of Z is provided in Table 4.2.7.

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† “Principles of Polymer Systems”, F. Rodriguez, Hemisphere Publishing Company (Chapter 10).  
†† This is conservative with respect to prior cask industry practice, which has historically utilized higher emissivities. For example, a higher emissivity for painted surfaces ( $\epsilon = 0.95$ ) is used in the previously licensed TN-32 cask TSAR (Docket 72-1021).

Table 4.2.1

SUMMARY OF HI-STORM SYSTEM MATERIALS  
THERMAL PROPERTY REFERENCES

Material	Emissivity	Conductivity	Density	Heat Capacity
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	EPRI [4.2.3]	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
UO <sub>2</sub>	Not Used	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Carbon Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Boral <sup>†</sup>	Not Used	Test Data	Test Data	Test Data
Holtite-A <sup>††</sup>	Not Used	Lower Bound Value Used	Not Used	Not Used
Concrete	Not Used	Marks' [4.2.1]	Marks' [4.2.1]	Handbook [4.2.2]
Lead	Not Used	Handbook [4.2.2]	Handbook [4.2.2]	Handbook [4.2.2]
Water	Not Used	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
Aluminum Alloy 1100 (Optional Heat Conduction Elements)	Handbook [4.2.2]	ASME [4.2.8]	ASME [4.2.8]	ASME [4.2.8]
<i>METAMIC</i> <sup>§</sup>	<i>Not Used</i>	<i>Test Data</i>	<i>Test Data</i>	<i>Test Data</i>

<sup>†</sup> AAR Structures Boral thermophysical test data.

<sup>††</sup> ~~From neutron shield manufacturer's data [1.2.11].~~

<sup>§</sup> Test data provided by METAMIC Inc.

Table 4.2.2

SUMMARY OF HI-STORM SYSTEM MATERIALS  
THERMAL CONDUCTIVITY DATA

<b>Material</b>	<b>@ 200°F (Btu/ft-hr-°F)</b>	<b>@ 450°F (Btu/ft-hr-°F)</b>	<b>@ 700°F (Btu/ft-hr-°F)</b>
Helium	0.0976	0.1289	0.1575
Air**	0.0173	0.0225	0.0272
Alloy X	8.4	9.8	11.0
Carbon Steel	24.4	23.9	22.4
Concrete††	1.05	1.05	1.05
Lead	19.4	17.9	16.9
Water	0.392	0.368	N/A

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\*\* At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F (at 32°F) and 0.0176 Btu/ft-hr-°F.at 212°F.

†† Assumed constant for the entire range of temperatures.

Table 4.2.3

SUMMARY OF FUEL ELEMENT COMPONENTS  
THERMAL CONDUCTIVITY DATA

<b>Zircaloy Cladding</b>		<b>Fuel (UO<sub>2</sub>)</b>	
Temperature (°F)	Conductivity (Btu/ft-hr-°F)	Temperature (°F)	Conductivity (Btu/ft-hr-°F)
392	8.28 <sup>†</sup>	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28 <sup>†</sup>

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<sup>†</sup> Lowest values of conductivity used in the thermal analyses for conservatism.



Table 4.2.4

SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA

<b>Material</b>	<b>Emissivity</b>
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel	0.36
Carbon Steel	0.66
<del>Sandblasted Aluminum</del>	<del>0.40</del>

Note: The emissivity of a metal surface is a function of the surface finish. In general, oxidation of a metal surface increases the emissivity. As stated in Marks' Standard Handbook for Mechanical Engineers: "Unless extraordinary pains are taken to prevent oxidation, however, a metallic surface may exhibit several times the emittance or absorptance of a polished specimen." This general statement is substantiated with a review of tabulated emissivity data from several standard references. These comparisons show that oxidized metal surfaces do indeed have higher emissivities than clean surfaces.

Table 4.2.5

## DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY

Material	Density (lbm/ft <sup>3</sup> )	Heat Capacity (Btu/lbm-°F)
Helium	(Ideal Gas Law)	1.24
Zircaloy	409	0.0728
Fuel (UO <sub>2</sub> )	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Boral	154.7	0.13
Concrete	142 <sup>†</sup>	0.156
Lead	710	0.031
Water	62.4	0.999
Aluminum Alloy 1100 (Optional Heat Conduction Elements)	169.9	0.23
<i>METAMIC</i>	<i>163.4 – 166.6</i>	<i>0.22 – 0.29</i>

<sup>†</sup> A minimum allowable density for concrete is specified as 146 lb/ft<sup>3</sup> (HI-STORM Overpack Serial Numbers 1 through 7) and 155 lb/ft<sup>3</sup> (HI-STORM Overpack Serial Number 8 onward) in Appendix 1.D. For conservatism in transient heatup calculations, the density is understated a lower value is specified here.

Table 4.2.6

GASES VISCOSITY<sup>†</sup> VARIATION WITH TEMPERATURE

Temperature (°F)	Helium Viscosity (Micropoise) <sup>††</sup>	Temperature (°F)	Air Viscosity (Micropoise)
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	-	-
463.0	288.7	-	-
537.8	299.8	-	-
737.6	338.8	-	-

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† Obtained from Rohsenow and Hartnett [4.2.2].

†† This data is also provided in graphical form in Figure 4.2.2.

Table 4.2.7

VARIATION OF NATURAL CONVECTION PROPERTIES  
PARAMETER “Z” FOR AIR WITH TEMPERATURE<sup>†</sup>

Temperature (°F)	Z (ft <sup>-3</sup> °F <sup>-1</sup> )
40	2.1×10 <sup>6</sup>
140	9.0×10 <sup>5</sup>
240	4.6×10 <sup>5</sup>
340	2.6×10 <sup>5</sup>
440	1.5×10 <sup>5</sup>

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<sup>†</sup> Obtained from Jakob and Hawkins [4.2.9].

Table 4.2.8

BORAL COMPONENT MATERIALS<sup>†</sup>  
THERMAL CONDUCTIVITY DATA

<b>Temperature (°F)</b>	<b>B<sub>4</sub>C Core Conductivity (Btu/ft-hr-°F)</b>	<b>Aluminum Cladding Conductivity (Btu/ft-hr-°F)</b>
212	48.09	100.00
392	48.03	104.51
572	47.28	108.04
752	46.35	109.43

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<sup>†</sup> Both B<sub>4</sub>C and aluminum cladding thermal conductivity values are obtained from AAR Structures Boral thermophysical test data.

Table 4.2.9

~~[INTENTIONALLY DELETED] HEAT CONDUCTION ELEMENTS (ALUMINUM  
ALLOY 1100)  
THERMAL CONDUCTIVITY DATA~~

<b>Temperature (°F)</b>	<b>Conductivity (Btu/ft-hr-°F)</b>
100	131.8
200	128.5
300	126.2
400	124.5

FIGURES 4.2.1 and 4.2.2

[INTENTIONALLY DELETED]

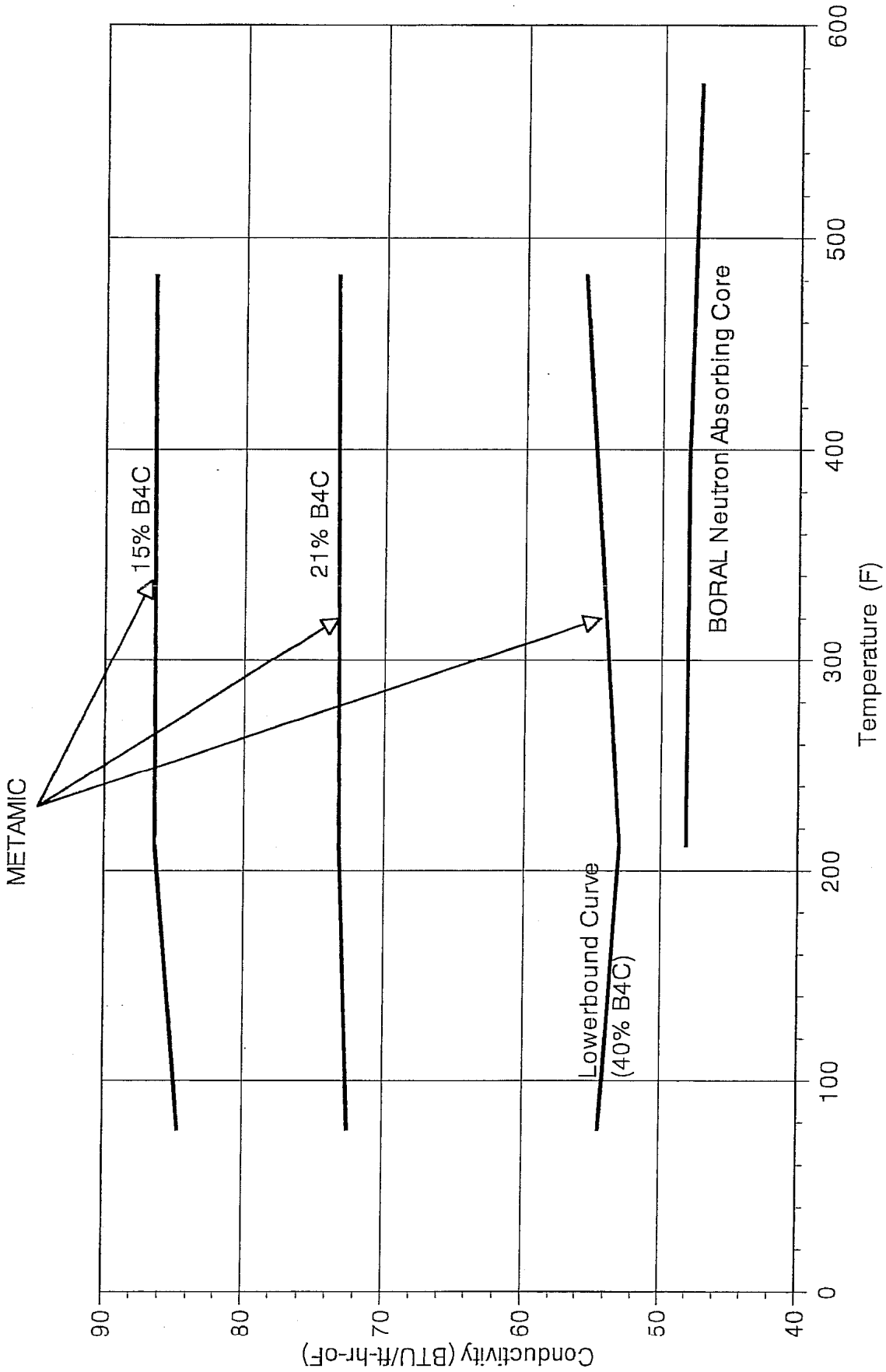


FIGURE 4.2.3: COMPARISON OF THERMAL CONDUCTIVITY OF METAMIC AND THE CERMET CORE OF A BORAL NEUTRON ABSORBER



### 4.3 SPECIFICATIONS FOR COMPONENTS

HI-STORM System materials and components designated as “Important to Safety” (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) which warrant special attention are summarized in Table 4.3.1. The neutron shielding ability of Holtite-A neutron shield material used in the HI-TRAC onsite-transfer cask overpack is ensured by demonstrating that the material exposure temperatures are maintained below the maximum allowable limit. Long-term integrity of SNF is ensured by the HI-STORM System thermal evaluation which performance that demonstrates that fuel cladding temperatures are maintained below design basis limits. Boral Neutron absorber materials used in MPC baskets for criticality control (a composite material composed of made from B<sub>4</sub>C and aluminum) is are stable up to 1000°F for short term and 850°F for long term dry storage<sup>†</sup>. However, for conservatism, a significantly lower maximum temperature limit is specified for thermal evaluation imposed. The overpack concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that American Concrete Institute (ACI) guidance on temperature limits (see Appendix 1.D) is followed-are not exceeded.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term off-normal and severe hypothetical accident conditions. The inherent mechanical stability characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of HI-STORM System thermal performance, material temperature limits for long-term normal, short-term operations, and off-normal and accident conditions under off-normal or hypothetical accident conditions, material temperature limits for short-duration events are provided in Table 4.3.1. In Table 4.3.1, ISG-11 [4.1.4] temperature limits are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

#### 4.3.1 Evaluation of Moderate Burnup Zircaloy Clad Fuel

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400 °C [4.3.8]. Upon cooling below a threshold temperature ( $T_p$ ), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature  $T_p$  are excessive. For moderate burnup fuel,  $T_p$  is conservatively estimated as 350 °C [4.3.8]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.8]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted in the HI-STORM FSAR for short-term operations with MBF fueled MPCs (see Table 4.3.1).

<sup>†</sup> B<sub>4</sub>C is a refractory material that is unaffected by high temperature (on the order of 1000 °F) and aluminum is solid at temperatures in excess of 1000 °F. AAR Advanced Structures Boral thermophysical test data.

Demonstration of fuel cladding integrity against the potential for degradation and gross rupture throughout the entire dry cask storage period is mandated by the Code of Federal Regulations (Part 72, Section 72.122(h)). The specific criteria required to establish fuel cladding integrity, set forth in NUREG-1536 (4.0, IV, 5&6) are:

- i. For each fuel type proposed for storage, the dry cask storage system should ensure a very low probability of cladding breach during long-term storage.
- ii. Fuel cladding damage resulting from creep cavitation should be limited to 15% of the original cladding cross-sectional area during dry storage.

Consistent with the NUREG-1536 criteria, the HI-STORM System is designed to preclude gross fuel cladding failures during the entire duration of storage. A method for establishing the peak cladding temperature limits in accordance with the diffusion-controlled cavity growth (DCCG) methodology was proposed by the Lawrence Livermore National Laboratory [4.3.5]. Recent NRC guidelines<sup>††</sup>, applicable for high burnup fuel (greater than 45,000 MWD/MTU), require that alternate methods be adopted for computing peak cladding temperature limits (see Appendix 4.A). For the FSAR request for approval for fuel burnups up to 45,000 MWD/MTU, the PNL-6189 [4.3.1] creep rupture criteria has been conservatively adopted in accord with the latest NRC guidelines so as to develop more restrictive permissible peak fuel cladding temperatures for the HI-STORM System. A discussion of the DCCG and PNL criteria for establishing allowable cladding temperatures is provided in the balance of the section.

#### 4.3.1.1 Cladding Temperature Limits (DCCG Criteria)

For SNF of a given age (decay time), the permissible peak cladding temperature is a direct function of the cladding hoop stress, which in turn depends on the radius-to-thickness ratio of the fuel rod and its internal pressure. The rod internal pressure  $P_i$  is a function of the maximum initial fill pressures (Tables 4.3.2 and 4.3.5) and fuel burnup-dependent fission gas release. The free rod volumes in the third column of Tables 4.3.2 and 4.3.5 are defined as free rod volumes in each fuel rod available for pressurization with fill gas. The free rod volume is the cumulative sum of the open top plenum space, the pellet-to-cladding annular space and the inter-pellet junction space. As a lower bound value of the free rod volume is conservative for cladding stress at operating temperatures, only the nominal gas plenum space is shown. The plenum length for miscellaneous BWR fuel assemblies is set to 12 inches. The radius-to-thickness ratio  $r^*$  is determined based on rod nominal dimension values (Tables 4.3.3 and 4.3.6), with consideration of maximum cladding thickness loss due to in-reactor oxidation, as reported by PNL [4.3.4].

The data presented in Tables 4.3.2 and 4.3.5 are combined with theoretical bounding fuel rod internal gas pressures from published technical sources [4.3.1 and 4.3.6], to absolutely ensure that bounding clad hoop stress values are used in the determination of gross cladding integrity. These

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<sup>††</sup> Interim Staff Guidance 11, "Storage of Spent Fuel Having Burnups in Excess of 45,000 MWD/MTU", USNRC.

bounding pressures are so large that they approach physical upper bounds for some fuel assemblies, as the corresponding hoop stresses approach the yield stress of zircaloy (approximately 172 MPa at 750°F [4.3.7]). The theoretical bounding rod internal pressure for PWR assemblies is compared, in Figure 4.3.1, to the published test data for assemblies from two different plants. From this figure, the large conservatism in the theoretical bounding pressure is evident.

These theoretical bounding pressures, from two sources, are provided below for PWR and BWR fuel:

- PWR: 2416 psia [4.3.1], 16 MPa (2320 psia) [4.3.6]
- BWR: 1094 psia [4.3.1], 70 atm (1029 psia) [4.3.6]

The coincident gas plenum temperatures reported in the PNL report [4.3.1] are 387°C for PWR assemblies and 311°C for BWR assemblies at reactor operating conditions. It can be seen in Figures 4.4.16 and 4.4.17 that the temperature distribution of gas in the fuel rods, a great bulk of which is located in the top gas plenum, is well below the in-core condition gas temperatures reported above (PWR fuel) and for the most part in the BWR fuel. In the interest of conservatism, no credit is taken for the substantially lower gas plenum temperatures that prevail during dry storage. Furthermore, the greater of the literature pressure data listed above is adopted for performing peak clad temperature limit calculations. The values utilized for  $P_i$  are 2416 psia for PWR<sup>†</sup> assemblies and 1094 psia for BWR assemblies.

By utilizing  $P_i$  and  $r^*$ , the cladding stress for various PWR fuel types is calculated from Lamé's formula and summarized in Table 4.3.3. For certain outlier fuel types (PWR), the stress calculations are provided in Table 4.3.9. An inspection of cladding stress data summarized in Tables 4.3.9 and 4.3.3 indicates 152.7 MPa as the theoretical bounding value of cladding stress ( $\sigma_{max}$ ) for the PWR SNF. Corresponding fill gas data and calculations of cladding stress for the various BWR SNF types are summarized in Tables 4.3.5 and 4.3.6, respectively. An inspection of the cladding stress data in Table 4.3.6 indicates that the theoretical bounding value of the cladding hoop stress for the BWR SNF is 72.7 MPa. The theoretical bounding values of  $\sigma_{max}$  for the array of PWR and BWR SNF types are thus 152.7 MPa and 72.7 MPa, respectively.

In this manner, the maximum conceivable values of cladding hoop stress are calculated for use in subsequent DCCG method calculations. As an additional conservatism, the peak fuel rod cladding hoop stresses are conservatively held constant throughout the dry storage period. In practice, the rod cladding hoop stresses are the maximum when the casks are initially loaded and monotonically decrease with the time-decreasing heat load and temperature. The Ideal Gas Law governs the decrease in pressure with decreasing temperature.

As stated earlier, the value of  $\sigma_{max}$  is required to establish the peak cladding temperature limit using the DCCG method. The DCCG model-based zircaloy cladding temperature limit computation, in accordance with the LLNL procedure [4.3.5], requires a solution to the following equation expressed

<sup>†</sup> Certain outlier fuels (Table 4.3.9) are stipulated to be below a postulated limiting rod pressure.

in terms of the area fraction of de-cohesion (A):

$$\int_{A_i}^{A_f} \frac{dA}{f(A)} = \int_{t_0}^{t_0+t_s} G(t) dt$$

where:

- $A_i$  = initial area fraction of de-cohesion
- $A_f$  = end of storage life area fraction of de-cohesion (limited to 0.15)
- $t_0$  = age of fuel prior to dry cask storage (years)
- $t_s$  = dry cask storage period (40 years)
- $f(A)$  = area fraction of de-cohesion function
- $G(t)$  = damage function

The term on the left hand side of this equation represents the area fraction of de-cohesion that occurs over the dry storage period. The term on the right hand side represents the cumulative damage over the same period. The area fraction of de-cohesion function and the damage function,  $f(A)$  and  $G(t)$  respectively, are:

$$f(A) = \frac{[1 - (\frac{A_i}{A})^{1/2}](1 - A)}{A^{1/2} [\frac{1}{2} \ln \frac{1}{A} - \frac{3}{4} + A(1 - \frac{A}{4})]}$$

$$G(t) = \frac{32 F_B^{3/2}(\alpha) \Omega \delta \sigma_\infty(t) D_{GB}[T(t)]}{3 \pi^{1/2} F_v(\alpha) K \lambda^3 T(t)}$$

where:

- $F_B(\alpha) = \pi \sin^2(\alpha)$
- $F_v(\alpha) = \frac{2\pi}{3} (2 - 3 \cos \alpha + \cos^3 \alpha)$
- $T(t)$  = time dependent peak cladding temperature
- $K$  = Boltzmann constant ( $1.38053 \times 10^{-23}$  J/K)

A discussion on the balance of parameters in the damage function  $G(t)$  is provided below.

### Cladding Hoop Stress ( $\sigma_\infty(t)$ )

The cladding hoop stress is principally dependent upon the specific fuel rod dimensions, initial fill rod pressure, time dependent storage temperature, and fuel burnup dependent fission gas release from the fuel pellets into the rod plenum space. The peak fuel rod pressure for various analyzed PWR and BWR fuel types at the start of the dry storage period are summarized in Tables 4.3.3 and 4.3.6. The highest peak rod stress among the various PWR and BWR fuel types, previously defined as  $\sigma_{max}$ , are conservatively applied as constant (time independent) cladding hoop stresses in the

~~DCCG model-based damage function.~~

Grain Boundary Cavity Dihedral Angle ( $\alpha$ )

~~The LLNL report [4.3.5] has determined the dihedral angle ( $\alpha$ ) for pure metals to be  $75^\circ$ . To account for possible non-ideal conditions, a conservatively lower  $\alpha$  equal to  $60^\circ$  is applied to the DCCG model.~~

Zirconium Atomic Volume ( $\Omega$ )

~~The zirconium atomic volume estimated from several literature sources as documented in the LLNL report [4.3.5] is in the range of  $2.31 \times 10^{-29} \text{ m}^3$  to  $3.37 \times 10^{-29} \text{ m}^3$ . In the interest of conservatism, the maximum estimated atomic volume equal to  $3.37 \times 10^{-29} \text{ m}^3$  is used for the analysis.~~

### Grain Boundary Thickness ( $\delta$ )

The LLNL report [4.3.5] has recommended a grain boundary thickness of three Burgers vectors to be adequate for the analysis. Thus,  $\delta = 3(3.23 \times 10^{-10}) = 9.69 \times 10^{-10}$  m is used in the analysis.

### Average Cavity Spacing ( $\lambda$ )

The type of nucleation mechanism and the density of nucleation sites control cavity spacing. The LLNL report [4.3.5] references an experimental study that found that the cavity spacing is in the range of  $10 \times 10^{-6}$  to  $20 \times 10^{-6}$  m. In the interest of conservatism, the minimum reported cavity spacing equal to  $10 \times 10^{-6}$  m is used in the analysis.

### Grain Boundary Diffusion Rate ( $D_{GB}$ )

Two grain boundary diffusion rate correlations for zirconium are reported in the LLNL report [4.3.5]. The two correlations provide diffusion rate estimates that are approximately two orders of magnitude apart from each other. Consequently, the more conservative correlation that provides a higher estimate of the grain boundary diffusion rate is used in the analysis. This more conservative correlation, yielding units of  $m^2/sec$ , is:

$$D_{GB} = 5.9 \times 10^{-6} \exp[-131,000/RT]$$

where R is the universal gas constant in J/mol $\times$ K units.

### Time Dependent Peak Cladding Temperature ( $T(t)$ )

The peak cladding temperature during long term storage is principally dependent upon the thermal heat load from the stored fuel assemblies, which is imposed on the cask. It is well established that the rate of radioactive decay in a fuel assembly exponentially attenuates with the age of fuel. Consequently, the peak cladding temperature during long term storage will also attenuate rapidly as a direct consequence of the heat load reduction with time, which is modeled using the data provided in USNRC Regulatory Guide 3.54 [4.3.3]. To confirm the applicability of the Reg. Guide 3.54 data, comparisons with the ORIGEN-S source term calculation results discussed in Chapter 5 of this FSAR were performed. Figures 4.3.2 and 4.3.3 present graphical comparisons of the decay heat versus decay time profiles from the Reg. Guide data with the profiles from the ORIGEN-S calculations. For the design basis maximum decay heat load (which is approached with 5 year old fuel), the Reg. Guide data agrees favorably with the ORIGEN-S calculation results. The Reg. Guide data is, in fact, slightly conservative with respect to the ORIGEN-S calculations.

It should be noted that the area fraction of de-cohesion function  $f(A)$  approaches zero in the limit as  $A \rightarrow A_i$ . Consequently, the mathematical singularity in the integral  $\int_{A_i}^A \frac{dA}{f(A)}$  is numerically accommodated by using an alternate form given below:

$$\int_{A_i}^A \frac{dA}{f(A)} = \text{Limit } \epsilon \rightarrow 0 \int_{A_i+\epsilon}^{A_f} \frac{A^{1/2} \left[ \frac{1}{2} \ln \frac{1}{A} - \frac{3}{4} + A \left(1 - \frac{A}{4}\right) \right] dA}{\left[1 - \left(\frac{A_i}{A}\right)^{1/2}\right] (1-A)}$$

The allowable area fraction of de-cohesion using  $A_i = 0.05$ ,  $\epsilon = 0.0001$ , and  $A_f = 0.15$  is determined to be equal to 0.15211.

This is consistent with an alternate form of the DCCG model reported in the PNL study [4.3.1, Appendix D] as reproduced below:

$$A_T = \int_0^{t_f} G(t) dt \leq 0.15$$

The cumulative damage  $G(t)$  can be evaluated as a function of the initial fuel cladding temperature and corresponding cladding stress, which are the two primary constituents of the damage function. The initial cladding hoop stress at a bounding storage temperature has already been determined. All other parameters in the  $G(t)$  function (except for the initial peak cladding temperature limit  $T_e$ ) have been defined as discussed previously in this section. The cumulative cladding damage experienced during the 40-year dry cask storage period is determined by integrating the  $G(t)$  function. The initial peak cladding temperature limit parameter  $T_e$  is iteratively adjusted to limit the cumulative damage to 15% as required by the NUREG-1536 Criterion (ii) discussed earlier in this section. The initial peak cladding temperature limits for the bounding PWR and BWR fuel assemblies are provided in Table 4.3.7.

#### 4.3.1.2 Permissible Cladding Temperatures (PNL Method)

In this subsection, the permissible peak clad temperature limits for the HI-STORM System are computed using the so-called “generic CSFM temperature limits” data provided in a PNL report [4.3.1]. The generic CSFM temperature limits, known to be more conservative than the previously discussed DCCG method, define the maximum permissible initial storage temperature ( $T_p$ ) of cladding as a function of initial cladding stress ( $\sigma_{max}$ ) and fuel age ( $\tau_f$ ) at the start of dry storage. The stress developed in cladding is a function of rod diameter-to-thickness ratio ( $d_e$ ) and the internal rod gas pressure ( $P_e$ ) which prevails during dry storage conditions. In the previous subsection, the W-14x14 and GE-7x7 fuel types were identified to have the highest  $d_e$  in the class of PWR† and

† Certain outlier fuels are excluded from this class as the cladding stress is bounded by the design basis W-14x14 fuel (Table 4.3.9).

BWR fuels, respectively. The cladding thickness data in Tables 4.3.3 and 4.3.6 is the corroded wall thickness after including maximum oxidation loss during reactor operation. The  $d_e$  for bounding PWR and BWR SNF is 18.3 and 19.3, respectively.

The cladding stress in a fuel rod is principally dependent upon the rod internal pressure  $P_o$ , which is postulated to reasonably bound rod pressures of SNF during dry storage. PNL [4.3.2] and EPRI [4.3.4] provide in-core irradiation rod pressures information which are theoretical upper bounds. For reference, they are provided herein in Subsection 4.3.1.1. Other robust sources† which authoritatively deal with this matter report peak rod pressures of 1600 psia (PWR) and 900 psia (BWR) during in-core irradiation. The conservatism in the in-core irradiation rod pressures for bounding rods pressure during dry cask storage is illustrated in Figure 4.3.1. From published test data on rods pressure measured from two different plants, the projected rods pressure in dry storage is significantly lower than the in-core irradiation pressure (~1350 psia for PWR). For computing permissible cladding temperatures for SNF storage in the HI-STORM System, a conservatively postulated  $P_o$  of 2000 psia (PWR) and 1000 psia (BWR) are employed in this work.

The dry storage rod pressure  $P_o$  for PWR and BWR types is postulated as 2000 psia and 1000 psia, respectively. Having obtained  $P_o$ , the cladding stress ( $\sigma_{max}$ ) is readily obtained by the product of  $P_o$  and  $d_e$  and dividing the result by 2 (Lame's formula). The cladding stress computed in this manner is 18,300 psi (126.1 MPa) and 9,650 psia (66.5 MPa) for PWR and BWR fuel, respectively. From the generic CSFM temperature limits table in the PNL report [4.3.1, page 3-19] and  $\sigma_{max}$ , the permissible peak clad temperature limit ( $T_p$ ) as a function of  $\tau_f$  is readily obtained. The  $T_p$  vs.  $\tau_f$  results for PWR and BWR fuel are presented in Table 4.3.7. The peak clad temperature limits (DCCG criteria) and permissible cladding temperature limits (PNL criteria) data are graphically depicted in Figure 4.3.4. The more restrictive results (PNL criteria) are applied to the HI-STORM System. In Table 4.3.8, permissible (PNL criteria) temperatures for an outlier fuel type (Dresden-1 thin clad) are evaluated at a conservatively bounding stress (94.1 MPa, Table 4.3.6). These temperatures are applicable to Low Heat Emitting (LHE) fuel evaluated in Subsection 4.4.1.1.13.

#### 4.3.2 Evaluation of Stainless Steel Clad Fuel

Approximately 2,200 PWR and BWR fuel assemblies stored in the United States were manufactured with stainless steel cladding. All stainless steel cladding materials are of the austenitic genre with the ASTM alloy compositions being principally type 304 and 348H. For long-term storage conditions, a recent EPRI/PNL study [4.3.4] recommends a 430°C (806°F) peak stainless steel cladding temperature limit. This temperature limit is substantially higher than the peak fuel cladding temperatures calculated for the HI-STORM System with design-basis maximum decay heat loads and zircaloy clad fuel (see Tables 4.4.9 and 4.4.10).

It is recognized that the peak cladding temperature of stainless fuel will differ from zircaloy clad fuel principally due to the following differences:

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† NRC SER for HI-STORM System (Docket 72-1014).



- i. — Differences in decay heat levels
- ii. — Differences in cladding emissivity
- iii. — Differences in cladding conductivity
- iv. — Differences in fuel rod array dimensions

The net planar thermal resistance of the equivalent homogenized axisymmetric MPC basket containing stainless steel clad fuel is greater than that with zircaloy clad fuel. The higher resistance arises principally from the significantly lower emissivity of the stainless steel cladding. This factor is, however, offset by significantly lower design basis heat loads prescribed for a HI-STORM System containing stainless steel clad fuel. A 20% (MPC 68, MPC 24, and MPC 24E) and 25% (MPC 32) or greater reduction in the design basis heat duty for stainless steel fuel (i.e., 20%–25% lower than zircaloy clad fuel) bounds the nominal percentage decrease in MPC basket effective thermal conductivity<sup>†</sup> (stainless steel fueled baskets are between 9% (MPC 68) to 25% (MPC 32) less conducting, as shown in Table 4.4.3). The design basis maximum allowable decay heat for MPCs fueled with stainless steel clad fuel are conservatively set to be 20% lower than zircaloy-fueled basket maximum heat load for MPC 24, MPC 24E, and MPC 68 (25% lower for MPC 32). Therefore, it is concluded that the peak cladding temperature for stainless steel clad fuel will be bounded by zircaloy clad fuel results. Consequently, in view of the conservative heat loads prescribed for stainless steel clad fuel, a separate thermal analysis to demonstrate the adequacy of stainless steel cladding integrity for storage in the HI-STORM System is not necessary.

#### 4.3.3 Short-Term Cladding Temperature Limit

For short-term durations, relatively high fuel cladding temperature limits have been historically accepted. For example, the Safety Analysis Report of the STC transport cask (Docket No. 71-9235), recently certified by the USNRC, permits 1200°F (approximately 649°C) as the maximum value of the peak cladding temperature,  $T_{max}$ , for transport of SNF with up to 45,000 MWD/MTU burnup. NUREG-1536 and PNL test data [4.3.2], limiting themselves to medium burnup levels (28,800 MWD/MTU), endorse a somewhat lower  $T_{max}$  ( $T_{max} = 570^\circ\text{C}$  or 1058°F). Based on the published industry test data, guidance in the literature, and analytical reasoning, we herein prescribe 570°C as the admissible value of  $T_{max}$  for SNF, with accumulated burnups up to 45,000 MWD/MTU, in the HI-STORM System.

A Brookhaven report written for EPRI [4.3.6] asserts that fuel cladding rupture becomes “virtually absent at stresses below about 200 MPa”. It can be readily deduced that the peak cladding stress for the limiting condition of 570°C cladding temperature will be below 200 MPa for the SNF burnup levels considered in this FSAR. Recalling that  $\sigma_{max} = 152.7$  MPa (Table 4.3.3) at a 387°C average rod gas temperature, the cladding circumferential stress  $\sigma_{peak}$  at 570°C is obtained by direct proportionality in absolute gas temperature:

$$\sigma_p = \sigma_{max} \times (570 + 273) / (387 + 273) = 195.0 \text{ MPa}$$

<sup>†</sup> The term “effective conductivity” of the fuel basket is defined in Section 4.4.1.

Therefore, a short-term fuel cladding temperature limit  $T_{\max} = 570^{\circ}\text{C}$  is considered safe to preclude fuel cladding failure. For fuel claddings which have been exposed to higher levels of in-core irradiation, the irradiation process progressively hardens the cladding material, making high burnup fuel less susceptible to stress-induced creep and fracture at these stress levels (up to 200 MPa). A recent high burnup fuel cladding integrity study by German researchers<sup>†</sup> corroborates this physical reasoning. In the German study, fuel rods with up to 64,000 MWD/MTU burnup were tested at substantially higher stresses (~400 MPa and 600 MPa) without cladding failure.

The EPRI report [4.3.6] cites experiments on fourteen irradiated Turkey Point Unit 3 rods carried out by Einziger et al.<sup>††</sup> in 1982 which showed no breach in cladding even after as much as 7% strain was accumulated in elevated temperatures lasting for 740–1,000 hours. Einziger's test data corroborates our selection of  $T_{\max} = 570^{\circ}\text{C}$  as the short duration limiting temperature.

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<sup>†</sup> “Short-time Creep and Rupture Tests on High Burnup Fuel Rod Cladding”, by W. Goll, E. Toscano and H. Spilker.

<sup>††</sup> “High Temperature Post Irradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods under Dry Storage Conditions,” by R.E. Einziger, S.D. Atkin, D.E. Stallrecht, and V.S. Pasupathi, *Nuclear Technology*, 57:65-80 (1982).

Table 4.3.1

HI-STORM SYSTEM MATERIAL TEMPERATURE LIMITS

Material	Normal Long-Term Temperature Limits [°F]	Short-Term Temperature Limits [°F]
<del>Zircaloy fuel</del> CSF cladding (zirconium alloys and stainless steel)	752 (Moderate <sup>†</sup> Burnup) See Table 4.3.7	Short-Term Operations 752 (HBF) 1058 (MBF) Off-Normal and Accident 1058
<del>Stainless steel fuel cladding</del>	806	1058
<del>Neutron Absorber Boraf</del> <sup>††</sup>	800	950
<del>Holtite-A</del> <sup>†††</sup>	300 N/A	300 350 (Short Term Operations)
<del>Concrete</del> <sup>‡</sup>	200	350
<del>Water</del>	N/A 307 <sup>††††</sup>	N/A 307 <sup>§</sup> (Short Term Operations) N/A (Off-Normal and Accident)

<sup>†</sup> ~~High burnup fuel storage limits are established in Appendix 4.A.~~

<sup>††</sup> ~~Based on AAR Structures Boraf thermophysical test data.~~

<sup>†††</sup> See Section 1.2.1.3.2.

<sup>‡</sup> *These values are applicable for concrete in the overpack body, overpack lid and overpack pedestal. As stated in Chapter 1 (Appendix 1.D, Table 1.D.1), these limits are compared to the through-thickness section average temperature.*

<sup>††††</sup> ~~Saturation temperature at HI-TRAC water jacket design pressure.~~

<sup>§</sup> *Saturation temperature at HI-TRAC water jacket design pressure.*

Tables 4.3.2 through 4.3.9 are intentionally deleted.

~~SUMMARY OF PWR ASSEMBLY RODS INITIAL GAS FILL DATA~~

Assembly Type	Rods Per Assembly	Free Rod Volume (in <sup>3</sup> )	Fill Pressure (psig) at 70°F	Fill Gas Volume at STP <sup>†</sup>	
				Per Rod (Liters)	Per Assembly (Liters)
W-14×14 Std.	179	0.67 <sup>††</sup>	0-460	0.845	151.2
W-15×15 Std.	204	0.67 <sup>††</sup>	0-475	0.633	129.1
W-17×17 Std.	264	0.59 <sup>††</sup>	275-500	0.666	175.8
B&W-15×15 Mark B	208	1.308	415	0.582	121.1
B&W-17×17 Mark C	264	0.819	435	0.381	100.6
CE-14×14 Std.	164	1.693	300-450	0.814	133.5
CE-16×16 Std.	220	1.411	300-450	0.678	149.2
B&W-15×15 Mark B-11	208	1.260	415	0.560	116.5
CE-14×14 (MP2)	176	1.728	300-450	0.831	146.2

† STP stands for standard temperature (0°C) and pressure (1-atmosphere).

†† Bounding low values verified from Holtec's proprietary information database.

Table 4.3.3

BOUNDING VALUES OF FUEL CLADDING STRESS FOR PWR SNF

	W- 14×14 Std.	W- 15×15 Std.	W- 17×17 Std.	B&W- 15×15 Mark B	B&W- 17×17 Mark C	CE- 14×14 Std.	CE- 16×16 Sys 80	CE-14×14 (MPa)
Fresh Fuel Rods O.D. (inch)	0.4220	0.422	0.374	0.430	0.379	0.440	0.382	0.440
End of Life Oxidation Thickness (inch) <sup>†</sup>	0.0027	0.0027	0.0027	0.0027	0.0027	0.0027	0.0027	0.0027
End of Life Rods O.D. (inch)	0.4166	0.4166	0.3686	0.4246	0.3736	0.4346	0.3766	0.4346
Rods I.D. (inch)	0.3734	0.373	0.329	0.377	0.331	0.384	0.332	0.388
Average Tube Diameter (inch)	0.3950	0.3948	0.3488	0.4008	0.3523	0.4093	0.3493	0.4113
Wall Thickness (inch)	0.0216	0.0218	0.0198	0.0238	0.0213	0.0253	0.0223	0.0233
Theoretical Bounding Rod Pressure (MPa gage) <sup>††</sup>	16.7	16.7	16.7	16.7	16.7	16.7	16.7	16.7
Bounding Cladding Stress (MPa)	152.7	151.2	147.1	140.6	138.1	135.0	130.8	147.4

† PNL 4835 [4.3.2] reported maximum cladding thickness loss due to in reactor oxidation.

†† PNL 6189 [4.3.1] data.

Table 4.3.4

~~INTENTIONALLY DELETED.~~

Table 4.3.5

## SUMMARY OF BWR ASSEMBLY RODS INITIAL GAS FILL DATA

Assembly Type	Rods Per Assembly	Free Rod Volume (in <sup>3</sup> )	Fill Pressure (psig) at 70°F	Fill Gas Volume at STP	
				Per Rod (liters)	Per Assembly (liters)
GE 7x7 (1966)	49	2.073	0-44.1 <sup>†</sup>	0.126	6.17
GE 7x7 (1968)	49	2.073	0-44.1	0.126	6.17
GE 7x7R	49	1.991	0-44.1	0.121	5.93
GE 8x8	60	1.504	0-44.1	0.0915	5.49
GE 8x8R	62	1.433	0-147 <sup>††</sup>	0.240	14.88
Exxon 9x9	79	1.323	58.8-88.2 <sup>†††</sup>	0.141	11.1
6x6 GE Dresden-1	36	2.304	58.8-88.2	0.245	8.82
6x6 Dresden-1 MOX	36	2.286	58.8-88.2	0.243	8.75
6x6 GE Humboldt Bay	36	2.346	58.8-88.2	0.250	9.0
7x7 GE Humboldt Bay	49	1.662	58.8-88.2	0.177	8.67
8x8 GE Dresden-1	64	1.235	58.8-88.2	0.131	8.38
8x8 SPC	63	1.615	58.8-88.2	0.172	10.8
9x9 SPC-2 water rods	79	1.248	58.8-88.2	0.133	10.5
9x9 SPC-1 water rod	80	1.248	58.8-88.2	0.133	10.6
9x9 GE11/GE13	74	1.389	58.8-88.2	0.150	11.1
9x9 Atrium 9B SPC	72	1.366	58.8-88.2	0.145	10.4
10x10 SVEA-96	96	1.022	58.8-88.2	0.109	10.5
10x10 GE12	92	1.167	58.8-88.2	0.124	11.4
6x6 Dresden-1	36	2.455	58.8-88.2	0.261	9.4
7x7 Oyster Creek	49	2.346	58.8-88.2	0.250	12.2
8x8 Oyster Creek	64	1.739	58.8-88.2	0.185	11.8
8x8 Quad+ Westinghouse	64	1.201	58.8-88.2	0.128	8.2
8x8 TVA Browns Ferry	61	1.686	58.8-88.2	0.179	10.9
9x9 SPC-5	76	1.249	58.8-88.2	0.133	10.1
ANF 8x8	62	1.61	58.8-88.2	0.172	10.7
ANF 9X (9x9)	72	1.249	58.8-88.2	0.133	9.6

Table 4.3.6

## BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNE

	GE-7x7 (1966)	GE-7x7 (1968)	GE-7x7R	GE-8x8	GE-8x8R	Exxon-9x9
Fresh Fuel Rods O.D. (inch)	0.563	0.570	0.563	0.493	0.483	0.42
End of Life Oxidation	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047

<sup>†</sup> Conservatively bounding for GE 7x7 (1966), GE 7x7 (1968), GE 7x7R and GE 8x8 (ORNL/TM-9591/V1 R1).

<sup>††</sup> Conservatively bounding initial fill pressure. ORNL/TM-9591/V1 R1 reports GE 8x8R pre-pressurized to 3 atm.

<sup>†††</sup> BWR fuel rods internal pressurization between 4 to 6 atm (PNL-4835).

Thickness (inch)						
End of Life Rods O.D. (inch)	0.5536	0.5606	0.5536	0.4836	0.4736	0.4106
Rods I.D. (inch)	0.499	0.499	0.489	0.425	0.419	0.36
Average Tube Diameter (inch)	0.5263	0.5298	0.5213	0.4543	0.4463	0.3853
Wall Thickness (inch)	0.0273	0.0308	0.0323	0.0293	0.0273	0.0253
Theoretical Bounding Rod Pressure (MPa gage) <sup>†</sup>	7.54	7.54	7.54	7.54	7.54	7.54
Bounding Cladding Stress (MPa)	72.7	64.8	60.8	58.5	61.6	57.4

<sup>†</sup> PNL-6189 [4.3.1] data.



Table 4.3.6 (continued)

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNE

	6x6 GE Dresden-1	6x6 MOX Dresden-1	6x6 GE Humboldt Bay	7x7 GE Humboldt Bay	8x8 GE Dresden-1	8x8 SPC
Fresh Fuel Rods O.D. (inch)	0.5645	0.5625	0.563	0.486	0.412	0.484
End of Life Oxidation Thickness (inch)	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047
End of Life Rods O.D. (inch)	0.5551	0.5531	0.5536	0.4766	0.4026	0.4746
Rods I.D. (inch)	0.4945	0.4925	0.499	0.4204	0.362	0.414
Average Tube Diameter (inch)	0.5248	0.5228	0.5263	0.4485	0.3813	0.4443
Wall Thickness (inch)	0.0303	0.0303	0.0273	0.0281	0.0203	0.0303
Theoretical Bounding Rod Pressure (MPa gage) <sup>†</sup>	7.54	7.54	7.54	7.54	7.54	7.54
Bounding Cladding Stress (MPa)	65.3	65.0	72.7	60.1	70.8	55.3

<sup>†</sup> PNL-6189 [4.3.1] data.

Table 4.3.6 (continued)

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNE

	9x9 SPC-2 water rods	9x9 SPC-1 water rod	9x9 GE 11/13	9x9 SPC Atrium 9B	10x10 SVEA- 96	10x10 GE12
Fresh Fuel Rods O.D. (inch)	0.424	0.423	0.44	0.433	0.379	0.404
End of Life Oxidation Thickness (inch)	0.0047	0.0047	0.0047	0.0047	0.0047	0.0047
End of Life Rods O.D. (inch)	0.4146	0.4136	0.4306	0.4236	0.3696	0.3946
Rods I.D. (inch)	0.364	0.364	0.384	0.3808	0.3294	0.352
Average Tube Diameter (inch)	0.3893	0.3888	0.4073	0.4022	0.3495	0.3733
Wall Thickness (inch)	0.0253	0.0248	0.0233	0.0214	0.0201	0.0213
Theoretical Bounding Rod Pressure (MPa gage) <sup>†</sup>	7.54	7.54	7.54	7.54	7.54	7.54
Bounding Cladding Stress (MPa)	58.0	59.1	65.9	70.9	65.6	66.1

<sup>†</sup> PNL 6189 [4.3.1] data.

Table 4.3.6 (continued)

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNF

	6x6 Dresden 1 <sup>†</sup> Thin Clad	7x7 Oyster Creek	8x8 Oyster Creek	8x8 Quad <sup>†</sup>
Fresh fuel Rods O.D. (inch)	0.5625	0.57	0.5015	0.4576
End of Life Oxidization Thin Clad (inch)	0.0047	0.0047	0.0047	0.0047
End of Life Rods O.D. (inch)	0.5531	0.5606	0.4921	0.4482
Rods I.D. (inch)	0.5105	0.499	0.4295	0.3996
Average Tube Diameter (inch)	0.5318	0.5298	0.4608	0.4239
Wall Thickness (inch)	0.0213	0.0308	0.0313	0.0243
Theoretical Boundary Rod Pressure (MPa gauge)	7.54	7.54	7.54	7.54
Bounding Cladding Stress (MPa)	94.1	64.5	55.5	65.8

<sup>†</sup> Outlier fuel type evaluated in Table 4.3.8.

Table 4.3.6 (continued)

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNF

	8x8 TVA Browns Ferry	9x9 SPC-5	ANF 8x8	ANF 9x (9x9)
O.D. (inch)	0.483	0.417	0.484	0.424
End of Life—Oxidation Thickness (inch)	0.0047	0.0047	0.0047	0.0047
End of Life Rods O.D. (inch)	0.4736	0.4076	0.4746	0.4146
Rods I.D. (inch)	0.423	0.364	0.414	0.364
Average Tube Diameter (inch)	0.4483	0.3858	0.4443	0.3893
Wall Thickness (inch)	0.0253	0.0218	0.0303	0.0253
Theoretical Bounding Rod Pressure (MPa)	7.54	7.54	7.54	7.54
Bounding Cladding Stress (MPa)	66.8	66.7	55.3	58.0

Table 4.3.7

ZIRCALOY CLADDING TEMPERATURE LIMITS AND PERMISSIBLE TEMPERATURES

Fuel Age (years)	PWR SNF (°C) [°F]		BWR SNF (°C) [°F]	
	DCCG Limit	Permissible PNL Limit	DCCG Limit	Permissible PNL Limit
5	419.4 [787]	366.0 [691]	440.2 [824]	393.2 [740]
6	416.7 [782]	358.0 [676]	436.2 [817]	377.9 [712]
7	397.0 [747]	335.0 [635]	416.4 [781]	353.7 [669]
10	379.4 [715]	329.6 [625]	398.9 [750]	347.9 [658]
15	370.2 [698]	323.2 [614]	390.2 [734]	341.1 [646]

Table 4.3.8

PERMISSIBLE TEMPERATURES FOR OUTLIER FUEL TYPES

<b>Fuel Age (Years)</b>	<b>6x6 Dresden-1 Thin Clad (°C) [°F] (BWR)</b>
5	383.7 [723]
6	370.9 [700]
7	347.7 [658]
10	342.1 [648]
15	334.9 [635]

Table 4.3.9

Bounding Cladding Stress for Outlier PWR Fuel

	B&W 15x15 Mark B-11
Fresh Fuel Rods O.D. (inch)	0.414
End of Life Oxidation Thickness (inch)	0.0027
End of Life Rods O.D. (inch)	0.4086
Rods I.D. (inch)	0.370
Average Rod Diameter (inch)	0.3893
Limiting Rod Pressure (MPa)	15 <sup>†</sup>
Bounding Cladding Stress (MPa)	151.3

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<sup>†</sup> Rod pressure to be limited to 2175 psia at 387°C gas plenum temperature.

FIGURES 4.3.1 THROUGH 4.3.4

[INTENTIONALLY DELETED]



## 4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

Under long-term storage conditions, the HI-STORM System (i.e., HI-STORM overpack and MPC) thermal evaluation is performed with the MPC cavity backfilled with helium. Thermal analysis results for the long-term storage scenarios are obtained and reported in this section.

### 4.4.1 Thermal Model

The MPC basket design consists of four distinct geometries to hold 24 or 32 PWR, or 68 BWR fuel assemblies. The basket is a matrix of square compartments designed to hold the fuel assemblies in a vertical position. The basket is a honeycomb structure of alloy steel (Alloy X) plates with full-length edge-welded intersections to form an integral basket configuration. All individual cell walls, except outer periphery cell walls in the MPC-68 and MPC-32, are provided with Boral neutron absorber sandwiched between the box wall and a stainless steel sheathing plate over the full length of the active fuel region.

The design basis decay heat generation (per PWR or BWR assembly) for long-term normal storage is specified in Table 2.1.26. The decay heat is conservatively considered to be non-uniformly distributed over the active fuel length based on the design basis axial burnup distributions provided in Chapter 2 (Table 2.1.11).

Transport of heat from the interior of the MPC to its outer surface is accomplished by a combination of conduction through the MPC basket metal grid structure, and conduction and radiation heat transfer in the relatively small helium gaps between the fuel assemblies and basket cell walls. Heat dissipation across the gap between the MPC basket periphery and the MPC shell is by a combination of helium conduction ~~and natural convection (by means of the “Rayleigh” effect)<sup>†</sup>~~ radiation across the gap ~~and conduction in the aluminum alloy 1100 heat conduction elements<sup>\*</sup>~~. MPC internal helium circulation is recognized in the thermal modeling analyses reported herein. Heat rejection from the outer surface of the MPC to the environment is primarily accomplished by convective heat transfer to a buoyancy driven airflow through the MPC-to-overpack annular gap. Inlet and outlet ducts in the overpack cylinder at its bottom and top, respectively, allow circulation of air through the annulus. A secondary heat rejection path from the outer surface of the MPC to the environment involves thermal radiation heat transfer across the annular gap, radial conduction through the overpack cylinder, and natural convection and thermal radiation from the outer surface of the overpack to the atmosphere.

#### 4.4.1.1 Analytical Model - General Remarks

Transport of heat from the heat generation region (fuel assemblies) to the outside environment (ambient air or ground) is analyzed broadly in terms of three interdependent thermal models.

1. The first model considers transport of heat from the fuel assembly to the basket cell walls. This model recognizes the combined effects of conduction (through helium) and radiation, and is essentially a finite element technology based update of the classical Wooton &

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<sup>†</sup> ~~Neglected in the thermal analyses for conservatism.~~

<sup>\*</sup> ~~Neglected in the thermal analyses for conservatism.~~

Epstein [4.4.1] (which considered radiative heat exchange between fuel rod surfaces) formulation.

2. The second model considers heat transport within an MPC cross section by conduction and radiation. The effective cross sectional thermal conductivity of the basket region, obtained from a combined fuel assembly/basket heat conduction-radiation model developed on ANSYS, is applied to an axisymmetric thermal model of the HI-STORM System on the FLUENT [4.1.2] code.
3. The third model deals with the transmission of heat from the MPC exterior surface to the external environment (heat sink). The upflowing air stream in the MPC/cask annulus extracts most of the heat from the external surface of the MPC, and a small amount of heat is radially deposited on the HI-STORM inner surface by conduction and radiation. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and radiative heat transfer mechanisms from the vertical (cylindrical shell) and top cover (flat) surfaces. The reduction in radiative heat exchange between cask outside vertical surfaces and ambient air, because of blockage from the neighboring casks arranged for normal storage at an ISFSI pad as described in Section 1.4, is recognized in the analysis. The overpack top plate is modeled as a heated surface in convective and radiative heat exchange with air and as a recipient of heat input through insolation. Insolation on the cask surfaces is based on 12-hour levels prescribed in 10CFR71, averaged over a 24-hour period, after accounting for partial blockage conditions on the sides of the overpack.

Subsections 4.4.1.1.1 through 4.4.1.1.9 contain a systematic description of the mathematical models devised to articulate the temperature field in the HI-STORM System. The description begins with the method to characterize the heat transfer behavior of the prismatic (square) opening referred to as the “fuel space” with a heat emitting fuel assembly situated in it. The methodology utilizes a finite element procedure to replace the heterogeneous SNF/fuel space region with an equivalent solid body having a well-defined temperature-dependent conductivity. In the following subsection, the method to replace the “composite” walls of the fuel basket cells with an equivalent “solid” wall is presented. Having created the mathematical equivalents for the SNF/fuel spaces and the fuel basket walls, the method to represent the MPC cylinder containing the fuel basket by an equivalent cylinder whose thermal conductivity is a function of the spatial location and coincident temperature is presented.

Following the approach of presenting descriptions starting from the inside and moving to the outer region of a cask, the next subsections present the mathematical model to simulate the overpack. Subsection 4.4.1.1.9 concludes the presentation with a description of how the different models for the specific regions within the HI-STORM System are assembled into the final FLUENT model.

#### 4.4.1.1.1 Overview of the Thermal Model

Thermal analysis of the HI-STORM System is performed by assuming that the system is subject to its maximum heat duty with each storage location occupied and with the heat generation rate in each stored fuel assembly equal to the design-basis maximum value. While the assumption of equal heat generation imputes a certain symmetry to the cask thermal problem, the thermal model must

incorporate three attributes of the physical problem to perform a rigorous analysis of a fully loaded cask:

- i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange is a nonlinear function of surface temperatures.
- ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profiles in the fuel assemblies.
- iii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the MPC is spatially distributed with the maximum values reached in the central core region.

It is clearly impractical to model every fuel rod in every stored fuel assembly explicitly. Instead, the cross section bounded by the inside of the storage cell, which surrounds the assemblage of fuel rods and the interstitial helium gas, is replaced with an “equivalent” square (solid) section characterized by an effective thermal conductivity. Figure 4.4.1 pictorially illustrates the homogenization concept. Further details of this procedure for determining the effective conductivity are presented in Subsection 4.4.1.1.2; it suffices to state here that the effective conductivity of the cell space will be a function of temperature because the radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) is a strong function of the temperatures of the participating bodies. Therefore, in effect, every storage cell location will have a different value of effective conductivity (depending on the coincident temperature) in the homogenized model. The temperature-dependent fuel assembly region effective conductivity is determined by a finite volume procedure, as described in Subsection 4.4.1.1.2.

In the next step of homogenization, a planar section of MPC is considered. With each storage cell inside space replaced with an equivalent solid square, the MPC cross section consists of a metallic gridwork (basket cell walls with each square cell space containing a solid fuel cell square of effective thermal conductivity, which is a function of temperature) circumscribed by a circular ring (MPC shell). There are ~~five~~four distinct materials in this section, namely the homogenized fuel cell squares, the Alloy X structural materials in the MPC (including ~~Boral~~neutron absorber sheathing), ~~neutron absorber Boral~~, ~~Alloy 1100 aluminum heat conduction elements~~, and helium gas. Each of the ~~five~~four constituent materials in this section has a different conductivity. It is emphasized that the conductivity of the homogenized fuel cells is a strong function of temperature.

In order to replace this thermally heterogeneous MPC section with an equivalent conduction-only region, resort to the finite element procedure is necessary. Because the rate of transport of heat within the MPC is influenced by radiation, which is a temperature-dependent effect, the equivalent conductivity of the MPC region must also be computed as a function of temperature. Finally, it is recognized that the MPC section consists of two discrete regions, namely, the basket region and the peripheral region. The peripheral region is the space between the peripheral storage cells and the MPC shell. This space is essentially full of helium surrounded by Alloy X plates ~~and optionally Alloy 1100 aluminum heat conduction elements~~. Accordingly, as illustrated in Figure 4.4.2 for MPC-68, the MPC cross section is replaced with two homogenized regions with temperature-dependent conductivities. In particular, the effective conductivity of the fuel cells is subsumed into

the equivalent conductivity of the basket cross section. The finite element procedure used to accomplish this is described in Subsection 4.4.1.1.4. The ANSYS finite element code is the vehicle for all modeling efforts described in the foregoing.

In summary, appropriate finite-element models are used to replace the MPC cross section with an equivalent two-region homogeneous conduction lamina whose local conductivity is a known function of coincident absolute temperature. Thus, the MPC cylinder containing discrete fuel assemblies, helium, ~~Boron~~ *neutron absorber* and Alloy X, is replaced with a right circular cylinder whose material conductivity will vary with radial and axial position as a function of the coincident temperature. Finally, HI-STORM is simulated as a radially symmetric structure with a buoyancy-induced flow in the annular space surrounding the heat generating MPC cylinder.

The thermal analysis procedure described above makes frequent use of equivalent thermal properties to ease the geometric modeling of the cask components. These equivalent properties are rigorously calculated values based on detailed evaluations of actual cask system geometries. All these calculations are performed conservatively to ensure a bounding representation of the cask system. This process, commonly referred to as submodeling, yields accurate (not approximate) results. Given the detailed nature of the submodeling process, experimental validation of the individual submodels is not necessary.

Internal circulation of helium in the sealed MPC is modeled as flow in a porous media in the fueled region containing the SNF (including top and bottom plenums). The basket-to-MPC shell clearance space is modeled as a helium filled radial gap to include the downcomer flow in the thermal model. The downcomer region, as illustrated in Figure 4.4.2, consists of an azimuthally varying gap formed by the square-celled basket outline and the cylindrical MPC shell. At the locations of closest approach a differential expansion gap (a small clearance on the order of 1/10 of an inch) is engineered to allow free thermal expansion of the basket. At the widest locations, the gaps are on the order of the fuel cell opening (~6" (BWR) and ~9" (PWR) MPCs). It is heuristically evident that heat dissipation by conduction is maximum at the closest approach locations (low thermal resistance path) and that convective heat transfer is highest at the widest gap locations (large downcomer flow). In the FLUENT thermal model, a radial gap that is large compared to the basket-to-shell clearance and small compared to the cell opening is used. As a relatively large gap penalizes heat dissipation by conduction and a small gap throttles convective flow, the use of a single gap in the FLUENT model understates both conduction and convection heat transfer in the downcomer region. ~~Heat dissipation by the inclusion of aluminum heat conduction elements, as stated earlier, is conservatively neglected in the HI-STORM thermal modeling.~~

The FLUENT thermal modeling methodology has been benchmarked with full-scale cask test data (EPRI TN-24P cask testing), as well as with PNNL's COBRA-SFS modeling of the HI-STORM System. The benchmarking work has been documented in a Holtec topical report HI-992252 ("Topical Report on the HI-STAR/HI-STORM Thermal Model and Its Benchmarking with Full-Size Cask Test Data").

In this manner, a loaded MPC standing upright on the ISFSI pad in a HI-STORM overpack is replaced with a right circular cylinder with spatially varying temperature-dependent conductivity. Heat is generated within the basket space in this cylinder in the manner of the prescribed axial

burnup distribution. In addition, heat is deposited from insolation on the external surface of the overpack. Under steady state conditions the total heat due to internal generation and insolation is dissipated from the outer cask surfaces by natural convection and thermal radiation to the ambient environment and from heating of upward flowing air in the annulus. Details of the elements of mathematical modeling are provided in the following.

#### 4.4.1.1.2 Fuel Region Effective Thermal Conductivity Calculation

Thermal properties of a large number of PWR and BWR fuel assembly configurations manufactured by the major fuel suppliers (i.e., Westinghouse, CE, B&W, and GE) have been evaluated for inclusion in the HI-STORM System thermal analysis. Bounding PWR and BWR fuel assembly configurations are determined using the simplified procedure described below. This is followed by the determination of temperature-dependent properties of the bounding PWR and BWR fuel assembly configurations to be used for cask thermal analysis using a finite volume (FLUENT) approach.

To determine which of the numerous PWR assembly types listed in Table 4.4.1 should be used in the thermal model for the PWR fuel baskets (MPC-24, MPC-24E, MPC-32), we must establish which assembly type has the maximum thermal resistance. The same determination must be made for the MPC-68, out of the menu of SNF types listed in Table 4.4.2. For this purpose, we utilize a simplified procedure that we describe below.

Each fuel assembly consists of a large array of fuel rods typically arranged on a square layout. Every fuel rod in this array is generating heat due to radioactive decay in the enclosed fuel pellets. There is a finite temperature difference required to transport heat from the innermost fuel rods to the storage cell walls. Heat transport within the fuel assembly is based on principles of conduction heat transfer combined with the highly conservative analytical model proposed by Wooton and Epstein [4.4.1]. The Wooton-Epstein model considers radiative heat exchange between individual fuel rod surfaces as a means to bound the hottest fuel rod cladding temperature.

Transport of heat energy within any cross section of a fuel assembly is due to a combination of radiative energy exchange and conduction through the helium gas that fills the interstices between the fuel rods in the array. With the assumption of uniform heat generation within any given horizontal cross section of a fuel assembly, the combined radiation and conduction heat transport effects result in the following heat flow equation:

$$Q = \sigma C_o F_\epsilon A [T_C^4 - T_B^4] + 13.5740 L K_{cs} [T_C - T_B]$$

where:

$F_\epsilon$  = Emissivity Factor

$$= \frac{1}{\left(\frac{1}{\epsilon_C} + \frac{1}{\epsilon_B} - 1\right)}$$

$\epsilon_C, \epsilon_B$  = emissivities of fuel cladding, fuel basket (see Table 4.2.4)

$C_o$  = Assembly Geometry Factor

$$= \frac{4N}{(N+1)^2} \text{ (when } N \text{ is odd)}$$

$$= \frac{4}{N+2} \text{ (when } N \text{ is even)}$$

$N$  = Number of rows or columns of rods arranged in a square array

$A$  = fuel assembly “box” heat transfer area =  $4 \times \text{width} \times \text{length}$

$L$  = fuel assembly length

$K_{cs}$  = fuel assembly constituent materials volume fraction weighted mixture conductivity

$T_C$  = hottest fuel cladding temperature ( $^{\circ}R$ )

$T_B$  = box temperature ( $^{\circ}R$ )

$Q$  = net radial heat transport from the assembly interior

$\sigma$  = Stefan-Boltzmann Constant ( $0.1714 \times 10^{-8} \text{ Btu/ft}^2\text{-hr-}^{\circ}R^4$ )

In the above heat flow equation, the first term is the Wooten-Epstein radiative heat flow contribution while the second term is the conduction heat transport contribution based on the classical solution to the temperature distribution problem inside a square shaped block with uniform heat generation [4.4.5]. The 13.574 factor in the conduction term of the equation is the shape factor for two-dimensional heat transfer in a square section. Planar fuel assembly heat transport by conduction occurs through a series of resistances formed by the interstitial helium fill gas, fuel cladding and enclosed fuel. An effective planar mixture conductivity is determined by a volume fraction weighted sum of the individual constituent material resistances. For BWR assemblies, this formulation is applied to the region inside the fuel channel. A second conduction and radiation model is applied between the channel and the fuel basket gap. These two models are combined, in series, to yield a total effective conductivity.

The effective conductivity of the fuel for several representative PWR and BWR assemblies is presented in Tables 4.4.1 and 4.4.2. At higher temperatures (approximately 450°F and above), the zircaloy clad fuel assemblies with the lowest effective thermal conductivities are the W-17×17 OFA (PWR) and the GE11-9×9 (BWR). A discussion of fuel assembly conductivities for some of the recent vintage 10×10 array and certain plant specific BWR fuel designs is presented near the end of this subsection. As noted in Table 4.4.2, the Dresden 1 (intact and damaged) fuel assemblies are excluded from consideration. The design basis decay heat load for Dresden-1 intact and damaged fuel (Table 2.1.7) is approximately 58% lower than the MPC-68 design-basis maximum heat load (Table 2.1.6). Examining Table 4.4.2, the effective conductivity of the damaged Dresden-1 fuel assembly in a damaged fuel container is approximately 40% lower than the bounding (GE-11 9×9) fuel assembly. Consequently, the fuel cladding temperatures in the HI-STORM System with Dresden-1 intact or damaged fuel assemblies will be bounded by design basis fuel cladding temperatures. Based on this simplified analysis, the W-17×17 OFA PWR and GE11-9×9 BWR fuel assemblies are determined to be the bounding configurations for analysis of zircaloy clad fuel at

design basis maximum heat loads. ~~As discussed in Section 4.3.1, stainless clad fuel assemblies with significantly lower decay heat emission characteristics are not deemed to be bounding.~~

For the purpose of determining axial flow resistance for inclusion of MPC thermosiphon effect in the HI-STORM system modeling, equivalent porous media parameters for the W-17x17OFA and GE11-9x9 fuels are computed. Theoretically bounding expansion and contraction loss factors are applied at the grid spacer locations to conservatively maximize flow resistance. As an additional measure of conservatism, the grids are modeled by postulating that they are formed using thick metal sheets which have the effect of artificially throttling flow. Heat transfer enhancement by grid spacers turbulence is conservatively ignored in the analysis.

Having established the governing (most resistive) PWR and BWR SNF types, we use a finite-volume code to determine the effective conductivities in a conservative manner. Detailed conduction-radiation finite-volume models of the bounding PWR and BWR fuel assemblies developed on the FLUENT code are shown in Figures 4.4.3 and 4.4.4, respectively. The PWR model was originally developed on the ANSYS code, which enables individual rod-to-rod and rod-to-basket wall view factor calculations to be performed using the AUX12 processor. Limitations of radiation modeling techniques implemented in ANSYS do not permit taking advantage of quarter symmetry of the fuel assembly geometry. Unacceptably long CPU time and large workspace requirements necessary for performing gray body radiation calculations for a complete fuel assembly geometry on ANSYS prompted the development of an alternate simplified model on the FLUENT code. The FLUENT model is benchmarked with the ANSYS model results for a Westinghouse 17x17 fuel assembly geometry for the case of black body radiation (emissivities = 1). The FLUENT model is found to yield conservative results in comparison to the ANSYS model for the “black” surface case. The FLUENT model benchmarked in this manner is used to solve the gray body radiation problem to provide the necessary results for determining the effective thermal conductivity of the governing PWR fuel assembly. The same modeling approach using FLUENT is then applied to the governing BWR fuel assembly, and the effective conductivity of GE-11 9x9 fuel determined.

The combined fuel rods-helium matrix is replaced by an equivalent homogeneous material that fills the basket opening by the following two-step procedure. In the first step, the FLUENT-based fuel assembly model is solved by applying equal heat generation per unit length to the individual fuel rods and a uniform boundary temperature along the basket cell opening inside periphery. The temperature difference between the peak cladding and boundary temperatures is used to determine an effective conductivity as described in the next step. For this purpose, we consider a two-dimensional cross section of a square shaped block with an edge length of  $2L$  and a uniform volumetric heat source ( $q_g$ ), cooled at the periphery with a uniform boundary temperature. Under the assumption of constant material thermal conductivity ( $K$ ), the temperature difference ( $\Delta T$ ) from the center of the cross section to the periphery is analytically given by [4.4.5]:

$$\Delta T = 0.29468 \frac{q_g L^2}{K}$$

This analytical formula is applied to determine the effective material conductivity from a known quantity of heat generation applied in the FLUENT model (smeared as a uniform heat source,  $q_g$ ) basket opening size and  $\Delta T$  calculated in the first step.

As discussed earlier, the effective fuel space conductivity must be a function of the temperature coordinate. The above two-step analysis is carried out for a number of reference temperatures. In this manner, the effective conductivity as a function of temperature is established.

In Table 4.4.5, 10×10 array type BWR fuel assembly conductivity results from a simplified analysis are presented to determine the most resistive fuel assembly in this class. The Atrium-10 fuel type is determined to be the most resistive in this class of fuel assemblies. A detailed finite-element model of this assembly type was developed to rigorously quantify the heat dissipation characteristics. The results of this study are presented in Table 4.4.6 and compared to the BWR bounding fuel assembly conductivity depicted in Figure 4.4.5. The results of this study demonstrate that the bounding fuel assembly conductivity is conservative with respect to the 10×10 class of BWR fuel assemblies.

Table 4.4.23 summarizes plant specific fuel types' effective conductivities. From these analytical results, SPC-5 is determined to be the most resistive fuel assembly in this group of fuel. A finite element model of the SPC-5 fuel assembly was developed to confirm that its in-plane heat dissipation characteristics are bounded from below by the Design Basis BWR fuel conductivities used in the HI-STORM thermal analysis.

Temperature-dependent effective conductivities of PWR and BWR design basis fuel assemblies (most resistive SNF types) are shown in Figure 4.4.5. The finite volume results are also compared to results reported from independent technical sources. From this comparison, it is readily apparent that FLUENT-based fuel assembly conductivities are conservative. The FLUENT computed values (not the published literature data) are used in the MPC thermal analysis presented in this document.

#### 4.4.1.1.3 Effective Thermal Conductivity of ~~Boral~~Neutron Absorber/Sheathing/Box Wall Sandwich

Each MPC basket cell wall (except the MPC-68 and MPC-32 outer periphery cell walls) is manufactured with a ~~Boral~~-neutron absorbing plate for criticality control. Each ~~Boral~~neutron absorber plate is sandwiched in a sheathing-to-basket wall pocket. A schematic of the “Box Wall - ~~Boral~~Neutron Absorber - Sheathing” sandwich geometry of an MPC basket is illustrated in Figure 4.4.6. During fabrication, a uniform normal pressure is applied to each “Box Wall - ~~Boral~~Neutron Absorber - Sheathing” sandwich in the assembly fixture during welding of the sheathing periphery on the box wall. This ensures adequate surface-to-surface contact for elimination of any macroscopic air gaps. The mean coefficient of linear expansion of the ~~Boral~~neutron absorber is higher than the thermal expansion coefficients of the basket and sheathing materials. Consequently, basket heat-up from the stored SNF will further ensure a tight fit of the ~~Boral~~neutron absorber plate in the sheathing-to-box pocket. The presence of small microscopic gaps due to less than perfect surface finish characteristics requires consideration of an interfacial contact resistance between the ~~Boral~~neutron absorber and box-sheathing surfaces. A conservative contact resistance resulting from a 2 mil Boral to pocket gap is applied in the analysis. In other words, no credit is taken for the interfacial pressure between ~~Boral~~neutron absorber and stainless plate/sheet stock produced by the fixturing and welding process.



Heat conduction properties of a composite “Box Wall - ~~Boral~~ *Neutron Absorber* - Sheathing” sandwich in the two principal basket cross sectional directions as illustrated in Figure 4.4.6 (i.e., lateral “out-of-plane” and longitudinal “in-plane”) are unequal. In the lateral direction, heat is transported across layers of sheathing, air-gap, *neutron absorber* ~~Boral~~ ( $B_4C$  and cladding layers) and box wall resistances that are essentially in series (except for the small helium filled end regions shown in Figure 4.4.7). Heat conduction in the longitudinal direction, in contrast, is through an array of essentially parallel resistances comprised of these several layers listed above. For the ANSYS based MPC basket thermal model, corresponding non-isotropic effective thermal conductivities in the two orthogonal sandwich directions are determined and applied in the analysis.

These non-isotropic conductivities are determined by constructing two-dimensional finite-element models of the composite “Box Wall — *Neutron Absorber* ~~Boral~~ - Sheathing” sandwich in ANSYS. A fixed temperature is applied to one edge of the model and a fixed heat flux is applied to the other edge, and the model is solved to obtain the average temperature of the fixed-flux edge. The equivalent thermal conductivity is then obtained using the resulting temperature difference across the sandwich as input to a one-dimensional Fourier equation as follows:

$$K_{\text{eff}} = \frac{q \times L}{T_h - T_c}$$

where:

- K<sub>eff</sub> = effective thermal conductivity
- q = heat flux applied in the ANSYS model
- L = ANSYS model heat transfer path length
- T<sub>h</sub> = ANSYS calculated average edge temperature
- T<sub>c</sub> = specified edge temperature

The heat transfer path length will vary, depending on the direction of transfer (i.e., in-plane or out-of-plane).

#### 4.4.1.1.4 Modeling of Basket Conductive Heat Transport

The total conduction heat rejection capability of a fuel basket is a combination of planar and axial contributions. These component contributions are calculated independently for each MPC basket design and then combined to obtain an equivalent isotropic thermal conductivity value.

The planar heat rejection capability of each MPC basket design (i.e., MPC-24, MPC-68, MPC-32 and MPC-24E) is evaluated by developing a thermal model of the combined fuel assemblies and composite basket walls geometry on the ANSYS finite element code. The ANSYS model includes a geometric layout of the basket structure in which the basket “Box Wall - ~~Boral~~ *Neutron Absorber* - Sheathing” sandwich is replaced by a “homogeneous wall” with an equivalent thermal conductivity. Since the thermal conductivity of the Alloy X material is a weakly varying function of temperature, the equivalent “homogeneous wall” must have a temperature-dependent effective conductivity. Similarly, as illustrated in Figure 4.4.7, the conductivities in the “in-plane” and “out-of-plane” directions of the equivalent “homogeneous wall” are different. Finally, as discussed earlier, the fuel assemblies and the surrounding basket cell openings are modeled as homogeneous heat generating

regions with an effective temperature dependent in-plane conductivity. The methodology used to reduce the heterogeneous MPC basket - fuel assemblage to an equivalent homogeneous region with effective thermal properties is discussed in the following.

Consider a cylinder of height,  $L$ , and radius,  $r_o$ , with a uniform volumetric heat source term,  $q_g$ , insulated top and bottom faces, and its cylindrical boundary maintained at a uniform temperature,  $T_c$ . The maximum centerline temperature ( $T_h$ ) to boundary temperature difference is readily obtained from classical one-dimensional conduction relationships (for the case of a conducting region with uniform heat generation and a constant thermal conductivity  $K_s$ ):

$$(T_h - T_c) = q_g r_o^2 / (4 K_s)$$

Noting that the total heat generated in the cylinder ( $Q_t$ ) is  $\pi r_o^2 L q_g$ , the above temperature rise formula can be reduced to the following simplified form in terms of total heat generation per unit length ( $Q_t/L$ ):

$$(T_h - T_c) = (Q_t / L) / (4 \pi K_s)$$

This simple analytical approach is employed to determine an effective basket cross-sectional conductivity by applying an equivalence between the ANSYS finite element model of the basket and the analytical case. The equivalence principle employed in the thermal analysis is depicted in Figure 4.4.2. The 2-dimensional ANSYS finite element model of the MPC basket is solved by applying a uniform heat generation per unit length in each basket cell region (depicted as Zone 1 in Figure 4.4.2) and a constant basket periphery boundary temperature,  $T_c$ . Noting that the basket region with uniformly distributed heat sources and a constant boundary temperature is equivalent to the analytical case of a cylinder with uniform volumetric heat source discussed earlier, an effective MPC basket conductivity ( $K_{eff}$ ) is readily derived from the analytical formula and ANSYS solution leading to the following relationship:

$$K_{eff} = N (Q_f' / L) / (4 \pi [T_h' - T_c'])$$

where:

$N$  = number of fuel assemblies

$(Q_f' / L)$  = per fuel assembly heat generation per unit length applied in ANSYS model

$T_h'$  = peak basket cross-section temperature from ANSYS model

Cross sectional views of MPC basket ANSYS models are depicted in Figures 4.4.9 and 4.4.10. ~~Notice that many of the basket supports and all shims have been conservatively neglected in the models. This conservative geometry simplification, coupled with the conservative neglect of thermal expansion that would minimize the gaps, yields conservative gap thermal resistances.~~ Temperature-dependent equivalent thermal conductivities of the fuel regions and composite basket walls, as determined from analysis procedures described earlier, are applied to the ANSYS model. The planar ANSYS conduction model is solved by applying a constant basket periphery temperature with uniform heat generation in the fuel region. The equivalent planar thermal conductivity values are lower bound values because, among other elements of conservatism, the effective conductivity of the most resistive SNF types (Tables 4.4.1 and 4.4.2) is used in the MPC finite element simulations.

The basket in-plane conductivities are computed for intact fuel storage and containerized fuel stored in Damaged Fuel Containers (DFCs). The MPC-24E is provided with four enlarged cells designated for storing damaged fuel. The MPC-68 has sixteen peripheral locations for damaged fuel storage in generic DFC designs. *The MPC-32 has eight peripheral locations for damaged fuel storage in generic DFC designs.* As a substantial fraction of the basket cells are occupied by intact fuel, the overall effect of DFC fuel storage on the basket heat dissipation rate is quite small. Including the effect of reduced conductivity of the DFC cells in MPC-24E, the basket conductivity is computed to drop slightly (~0.6%). In a bounding calculation in which all cells of MPC-68 are assumed occupied by fuel in DFC, the basket conductivity drops by about 5%. *In a bounding calculation in which all cells of an MPC-32 are assumed occupied by fuel in DFCs, the basket conductivity drops by about 17%.* Conservatively, assuming 95% of intact fuel basket heat load adequately covers damaged fuel storage in the MPC-24E and MPC-68 and assuming 80% of intact fuel basket heat load adequately covers damaged fuel storage in the MPC-32.

The axial heat rejection capability of each MPC basket design is determined by calculating the area occupied by each material in a fuel basket cross-section, multiplying by the corresponding material thermal conductivity, summing the products and dividing by the total fuel basket cross-sectional area. In accordance with NUREG-1536 guidelines, the only portion of the fuel assemblies credited in these calculations is the fuel rod cladding (*i.e., the contribution of fuel pellets to axial heat conduction is ignored*).

Having obtained planar and axial effective thermal conductivity contributions as described above, an equivalent isotropic thermal conductivity that yields the same overall heat transfer can be obtained. Two-dimensional conduction heat transfer in relatively short cylinders cannot be readily evaluated analytically, so an alternate approach is used herein.

Instead of computing precise isotropic conductivities, an RMS function of the planar and axial effective thermal conductivity values is used as follows:

$$k_{\text{iso}} = \sqrt{\frac{k_{\text{rad}}^2 + k_{\text{ax}}^2}{2}}$$

where:

$k_{\text{iso}}$  = equivalent isotropic thermal conductivity

$k_{\text{rad}}$  = equivalent planar thermal conductivity

$k_{\text{ax}}$  = equivalent axial thermal conductivity

This formulation has been benchmarked for specific application to the MPC basket designs and found to yield conservative equivalent isotropic thermal conductivities and, subsequently, conservative temperature results from subsequent thermal analyses.

Table 4.4.3 summarizes the isotropic MPC basket thermal conductivity values used in the subsequent cask thermal modeling. It should be noted that the isotropic conductivities calculated as described above are actually higher than those reported in Table 4.4.3, imparting additional conservatism to the subsequent calculations.

#### 4.4.1.1.5 Subsection Intentionally Deleted Heat Transfer in MPC Basket Peripheral Region

Both of the MPC designs for storing PWR or BWR fuel are provided with relatively large regions, formed between the relatively cooler MPC shell and hot basket peripheral panels, filled with helium gas. Heat transfer in these helium-filled regions corresponds to the classical case of heat transfer in a differentially heated closed cavity. Many investigators, including Eckert and Carlson (Int. J. Heat Mass Transfer, vol. 2, p. 106, 1961) and Elder (J. Fluid Mech., vol. 23, p. 77, 1965) have performed experimental studies of this arrangement. The peripheral region between the basket and MPC inner surface is simulated as a tall fluid-filled cavity of height H formed between two differentially heated surfaces ( $\Delta T$ ) separated by a small distance L. In a closed cavity, an exchange of hot and cold fluids occurs near the top and bottom ends of the cavity, resulting in a net transport of heat across the gap. The rate of heat transfer across the cavity is characterized by a Rayleigh number,  $Ra_L$ , defined as:

$$Ra_L = \frac{C_p \rho^2 g \beta \Delta T L^3}{\mu K}$$

where:

- $C_p$  = fluid heat capacity
- $\rho$  = fluid density
- $g$  = acceleration due to gravity
- $\beta$  = coefficient of thermal expansion (equal to reciprocal of absolute temperature for gases)
- $\Delta T$  = temperature difference between the hot and cold surfaces
- $L$  = spacing between the hot and cold surfaces
- $\mu$  = fluid viscosity
- $K$  = fluid conductivity

Hewitt et al. [4.4.6] recommends the following Nusselt number correlation for heat transport in tall cavities:

$$Nu_L = 0.42 Ra_L^{1/4} Pr^{0.012} \left(\frac{H}{L}\right)^{-0.3}$$

where Pr is the Prandtl number of the cavity fill gas.

A Nusselt number of unity implies heat transfer by fluid conduction only, while a higher than unity Nusselt number is due to the “Rayleigh” effect which monotonically increases with increasing Rayleigh number. Nusselt numbers applicable to helium-filled PWR and BWR fueled HI-STORM MPC peripheral voids used in the original licensing analysis are provided in Table 4.4.4. For conservatism, however, the contribution of the Rayleigh effect is ignored in the thermal model of the MPC.

#### 4.4.1.1.6 Subsection Intentionally Deleted Effective Thermal Conductivity of MPC Basket to Shell Aluminum Heat Conduction Elements

As shown in HI-STORM System MPC drawings in Section 1.5, an option for insertion of full-length heat conduction elements fabricated from thin aluminum Alloy 1100 sheet metal is shown in the

MPC design drawings. Due to the high thermal conductivity of aluminum Alloy 1100 (about 15 times that of Alloy X), a significant rate of net heat transfer is possible along thin plates. Figure 4.4.11 shows the mathematical idealization of a typical conduction element inserted in a basket periphery panel to MPC shell space. The aluminum heat conduction element is shown to cover the MPC basket Alloy X peripheral panel and MPC shell (Regions I and III depicted in Figure 4.4.11) surfaces along the full length of the basket except for isolated locations where fitup or interference with other parts precludes complete basket coverage. Heat transport to and from the aluminum heat conduction element is conservatively postulated to occur across a thin helium gap as shown in the figure (i.e., no credit is taken for contact between the aluminum heat conduction element and the Alloy X fuel basket). Aluminum surfaces inside the hollow region are sandblasted prior to fabrication to result in a rough surface finish which has a significantly higher emissivity compared to smooth surfaces of rolled aluminum. The untreated aluminum surfaces directly facing Alloy X panels have a smooth finish to minimize contact resistance.

Net heat transfer resistance from the hot basket periphery panel to the relatively cooler MPC shell along the aluminum heat conduction element pathway is a sum of three individual resistances, in regions labeled I, II, and III in Figure 4.4.11. In Region I, heat is transported from the basket to the aluminum heat conduction element surface directly facing the basket panel across a thin helium resistance gap. Longitudinal transport of heat (in the z direction) in the aluminum plate (in Region I) will result in an axially non uniform temperature distribution. Longitudinal one dimensional heat transfer in the Region I aluminum plate was analytically formulated to result in the following ordinary differential equation for the non uniform temperature distribution:

$$t K_{Al} \frac{\partial^2 T}{\partial z^2} = \frac{K_{He}}{h} (T_h - T)$$

Boundary Conditions

$$\frac{\partial T}{\partial z} = 0 \text{ at } z = 0$$

$$T = T_h' \text{ at } z = P$$

where (see Figure 4.4.11):

- T(z) = non uniform aluminum metal temperature distribution
- t = heat conduction element thickness
- K<sub>Al</sub> = heat conduction element conductivity
- K<sub>He</sub> = helium conductivity
- h = helium gap thickness
- T<sub>h</sub> = hot basket temperature
- T<sub>h</sub>' = heat conduction element Region I boundary temperature at z = P
- P = heat conduction element Region I length

Solution of this ordinary differential equation subject to the imposed boundary condition is:

$$(T_h - T) = (T_h - T_h') \frac{\left[ e^{\frac{z}{\sqrt{\alpha}}} + e^{-\frac{z}{\sqrt{\alpha}}} \right]}{\left[ e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}} \right]}$$

where  $\alpha$  is a dimensional parameter equal to  $(h \times t \times K_{Al} / K_{He})$ . The net heat transfer ( $Q_I$ ) across the Region I helium gap can be determined by the following integrated heat flux to a heat conduction element of length  $L$  as:

$$Q_I = \int_0^P \frac{K_{He}}{h} (T_h - T) (L) dz$$

Substituting the analytical temperature distribution result obtained in Equation c, the following expression for net heat transfer is obtained:

$$Q_I = \frac{K_{He} L \sqrt{\alpha}}{h} \left( 1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right) (T_h - T_{h'})$$

Based on this result, an expression for Region I resistance is obtained as shown below:

$$R_I = \frac{T_h - T_{h'}}{Q_I} = \frac{h}{K_{He} L \sqrt{\alpha}} \left( 1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right)^{-1}$$

The Region II resistance expression can be developed from the following net heat transfer equation in the vertical leg of the conduction element as shown below:

$$Q_{II} = \frac{K_{Al} L t}{W} (T_{h'} - T_c')$$

where  $W$  is the conduction element Region II length:

$$R_{II} = \frac{T_{h'} - T_c'}{Q_{II}} = \frac{W}{K_{Al} L t}$$

Similarly, a Region III resistance expression can be analytically determined as shown below:

$$R_{III} = \frac{(T_c' - T_c)}{Q_{III}} = \frac{h}{K_{He} L \sqrt{\alpha}} \left( 1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right)^{-1}$$

This completes the analysis for the total thermal resistance attributable to the heat conduction elements, which is equal to the sum of the three individual resistances. The total heat conduction element resistance is smeared across the basket to MPC shell region as an effective uniform annular gap conductivity (see Figure 4.4.2). We note that heat transport along the conduction elements is an independent conduction path in parallel with conduction and radiation mechanisms in the large helium gaps. Helium conduction and radiation in the MPC basket to MPC shell peripheral gaps is accounted for separately in the ANSYS models for the MPCs, described earlier. Therefore, the net conductivity of the MPC basket to MPC shell peripheral gap region is the sum of the heat conduction elements effective conductivity and the helium gap conduction radiation effective conductivity. For conservatism, however, the contribution of the heat conduction elements is ignored in the HI-STORM thermal analyses.

#### 4.4.1.1.7 Annulus Air Flow and Heat Exchange

The HI-STORM storage overpack is provided with four inlet ducts at the bottom and four outlet ducts at the top. The ducts are provided to enable relatively cooler ambient air to flow through the annular gap between the MPC and storage overpack in the manner of a classical “chimney”. Hot air is vented from the top outlet ducts to the ambient environment. Buoyancy forces induced by density differences between the ambient air and the heated air column in the MPC-to-overpack annulus sustain airflow through the annulus.

In contrast to a classical chimney, however, the heat input to the HI-STORM annulus air does not occur at the bottom of the stack. Rather, the annulus air picks up heat from the lateral surface of the MPC shell as it flows upwards. The height dependent heat absorption by the annulus air must be properly accounted for to ensure that the buoyant term in the Bernoulli equation is not overstated making the solution unconservative. To fix ideas, consider two cases of stack heat input; Case A where the heat input to the rising air is all at the bottom (the “fireplace” scenario), and Case B, where the heat input is uniform along the entire height (more representative of the ventilated cask conditions). In both cases, we will assume that the air obeys the perfect gas law; i.e., at constant pressure,  $\rho = C/T$  where  $\rho$  and  $T$  are the density and the absolute temperature of the air and  $C$  is a constant.

##### Case A: Entire Heat Input at the Bottom

In a stack of height  $H$ , where the temperature of the air is raised from  $T_i$  to  $T_o$  at the bottom (Figure 4.4.12; Case A), the net fluid “head”  $p_1$  is given by:

$$p_1 = \rho_i H - \rho_o H$$

$\rho_i$  and  $\rho_o$  are the densities of air corresponding to absolute temperatures  $T_i$  and  $T_o$ , respectively.

Since  $\rho_i = \frac{C}{T_i}$  and  $\rho_o = \frac{C}{T_o}$ , we have:

$$p_1 = CH \left( \frac{1}{T_i} - \frac{1}{T_o} \right)$$

or

$$p_1 = \frac{CH \Delta T}{T_i T_o}$$

where:  $\Delta T = T_o - T_i$

Let  $\Delta T \ll T_i$ , then we can write:

$$\begin{aligned}\frac{1}{T_o} &= \frac{1}{T_i \left(1 + \frac{\Delta T}{T_i}\right)} \\ &= \frac{1}{T_i} \left[ 1 - \frac{\Delta T}{T_i} + \dots \right]\end{aligned}$$

Substituting in the above we have:

$$p_1 = \frac{CH}{T_i} \delta (1 - \delta + \dots)$$

where  $\delta = \frac{\Delta T}{T_i}$  (dimensionless temperature rise)

or  $p_1 = \rho_i H \delta - O(\delta^2)$ .

### Case B: Uniform Heat Input

In this case, the temperature of air rises linearly from  $T_i$  at the bottom to  $T_o$  at the top (Figure 4.4.12; Case B):

$$T_o = T_i + \zeta h ; 0 \leq h \leq H$$

where:

$$\zeta = \frac{T_o - T_i}{H} = \frac{\delta T_i}{H}$$

The total buoyant head, in this case, is given by:

$$\begin{aligned}p_2 &= \rho_i H - \int_0^H \rho \, dh \\ &= \rho_i H - C \int_0^H \frac{1}{T} \, dh \\ &= \rho_i H - C \int_0^H \frac{dh}{(T_i + \zeta h)} \\ &= \rho_i H - \frac{C}{\zeta} \ln(1 + \delta)\end{aligned}$$

Using the logarithmic expansion relationship and simplifying we have:

$$p_2 = \frac{\rho_i H \delta}{2} - O(\Delta^2)$$



Neglecting terms of higher order, we conclude that  $p_2$  is only 50% of  $p_1$ , i.e., the buoyancy driver in the case of uniformly distributed heat input to the air is half of the value if the heat were all added at the bottom.

In the case of HI-STORM, the axial heat input profile into the annulus air will depend on the temperature difference between the MPC cylindrical surface and the rising air along the height (Case C in Figure 4.4.12). The MPC surface temperature profile, of course, is a strong function of the axial decay heat generation profile in the SNF. Previous analyses show that the HI-STORM “chimney” is less than 50% as effective as a classical chimney. As we explain in Subsection 4.4.1.1.9, this fact is fully recognized in the global HI-STORM thermal model implementation of FLUENT.

#### 4.4.1.1.8 Determination of Solar Heat Input

The intensity of solar radiation incident on an exposed surface depends on a number of time varying terms. The solar heat flux strongly depends upon the time of the day as well as on latitude and day of the year. Also, the presence of clouds and other atmospheric conditions (dust, haze, etc.) can significantly attenuate solar intensity levels. Rapp [4.4.2] has discussed the influence of such factors in considerable detail.

Consistent with the guidelines in NUREG-1536 [4.4.10], solar input to the exposed surfaces of the HI-STORM overpack is determined based on 12-hour insolation levels recommended in 10CFR71 (averaged over a 24-hour period) and applied to the most adversely located cask after accounting for partial blockage of incident solar radiation on the lateral surface of the cask by surrounding casks. In reality, the lateral surfaces of the cask receive solar heat depending on the azimuthal orientation of the sun during the course of the day. In order to bound this heat input, the lateral surface of the cask is assumed to receive insolation input with the solar insolation applied horizontally into the cask array. The only reduction in the heat input to the lateral surface of the cask is due to partial blockage offered by the surrounding casks. In contrast to its lateral surface, the top surface of HI-STORM is fully exposed to insolation without any mitigation effects of blockage from other bodies. In order to calculate the view factor between the most adversely located HI-STORM system in the array and the environment, a conservative geometric simplification is used. The system is reduced to a concentric cylinder model, with the inner cylinder representing the HI-STORM unit being analyzed and the outer shell representing a reflecting boundary (no energy absorption).

Thus, the radius of the inner cylinder ( $R_i$ ) is the same as the outer radius of a HI-STORM overpack. The radius of the outer cylinder ( $R_o$ ) is set such that the rectangular space ascribed to a cask is preserved. This is further explained in the next subsection. It can be shown that the view factor from the outer cylinder to the inner cylinder ( $F_{o-i}$ ) is given by [4.4.3]:

$$F_{o-i} = \frac{1}{R} - \frac{1}{\pi R} \times \left[ \cos^{-1} \left( \frac{B}{A} \right) - \frac{1}{2L} \left\{ \sqrt{(A+2)^2 - (2R)^2} \times \cos^{-1} \left( \frac{B}{RA} \right) + B \sin^{-1} \left( \frac{1}{R} \right) - \frac{\pi A}{2} \right\} \right]$$

where:

$F_{o-i}$  = View Factor from the outer cylinder to the inner cylinder  
 $R$  = Outer Cylinder Radius to Inner Cylinder Radius Ratio ( $R_o/R_i$ )  
 $L$  = Overpack Height to Radius Ratio  
 $A = L^2 + R^2 - 1$   
 $B = L^2 - R^2 + 1$

Applying the theorem of reciprocity, the view factor ( $F_{i-a}$ ) from outer overpack surface, represented by the inner cylinder, to the ambient can be determined as:

$$F_{i-a} = 1 - F_{o-i} \frac{R_o}{R_i}$$

Finally, to bound the quantity of heat deposited onto the HI-STORM surface by insolation, the absorptivity of the cask surfaces is assumed to be unity.

#### 4.4.1.1.9 FLUENT Model for HI-STORM

In the preceding subsections, a series of analytical and numerical models to define the thermal characteristics of the various elements of the HI-STORM System are presented. The thermal modeling begins with the replacement of the Spent Nuclear Fuel (SNF) cross section and surrounding fuel cell space with a solid region with an equivalent conductivity. Since radiation is an important constituent of the heat transfer process in the SNF/storage cell space, and the rate of radiation heat transfer is a strong function of the surface temperatures, it is necessary to treat the equivalent region conductivity as a function of temperature. Because of the relatively large range of temperatures in a loaded HI-STORM System under the design basis heat loads, the effects of variation in the thermal conductivity of the Alloy X basket wall with temperature are included in the numerical analysis model. The presence of significant radiation effects in the storage cell spaces adds to the imperative to treat the equivalent storage cell lamina conductivity as temperature-dependent.

Numerical calculations and FLUENT finite-volume simulations have been performed to establish the equivalent thermal conductivity as a function of temperature for the limiting (thermally most resistive) BWR and PWR spent fuel types. Utilizing the most limiting SNF (established through a simplified analytical process for comparing conductivities) ensures that the numerical idealization for the fuel space effective conductivity is conservative for all non-limiting fuel types.

Having replaced the fuel spaces by solid square blocks with a temperature-dependent conductivity essentially renders the basket into a non-homogeneous three-dimensional solid where the non-homogeneity is introduced by the honeycomb basket structure composed of interlocking basket panels. The basket panels themselves are a composite of Alloy X cell wall, Boron neutron absorber, and Alloy X sheathing metal. A conservative approach to replace this composite section with an equivalent “solid wall” was described earlier.

In the next step, a planar section of the MPC is considered. The MPC contains a non-symmetric basket lamina wherein the equivalent fuel spaces are separated by the “equivalent” solid metal walls.

The space between the basket and the MPC, called the peripheral gap, is filled with helium gas. At this stage in the thermal analysis, the SNF/basket/MPC assemblage has been replaced with a two-zone (Figure 4.4.2) cylindrical solid whose thermal conductivity is a strong function of temperature.

The fuel assembly and MPC basket effective conductivity evaluations are performed for two distinct scenarios described earlier in this section. In the first scenario, the MPC cavity is backfilled with helium only. In the second scenario, gaseous fission products from a hypothetical rupture of 10% of the stored fuel rods dilute the backfill helium gas. As previously stated, thermal analysis results for both scenarios are obtained and reported in this section.

The thermal model for the HI-STORM overpack is prepared as a ~~three~~two-dimensional axisymmetric body. For this purpose, the hydraulic resistances of the inlet ducts and outlet ducts, respectively, are represented by equivalent axisymmetric porous media. Two overpack configurations are evaluated – HI-STORM 100 and a shorter variation (HI-STORM 100S) overpack. HI-STORM 100S features a smaller inlet duct-to-outlet duct separation and an optional enhanced gamma shield cross plate. Since the optional gammas shield cross plate flow resistance is bounding, the optional design was conservatively evaluated in the thermal analysis. The fuel cladding temperatures for MPC emplaced in a HI-STORM 100S overpack are confirmed to be bounded by the HI-STORM 100 System thermal model solution. Thus, separate table summaries for HI-STORM 100S overpack are not provided. The axial resistance to airflow in the MPC/overpack annulus (which includes longitudinal channels to “cushion” the stresses in the MPC structure during a postulated non-mechanistic tip-over event) is replaced by a hydraulically equivalent annulus. The surfaces of the ducts and annulus are assumed to have a relative roughness ( $\epsilon$ ) of 0.001. This value is appropriate for rough cast iron, wood stave and concrete pipes, and is bounding for smooth painted surfaces (all readily accessible internal and external HI-STORM overpack carbon steel surfaces are protected from corrosion by painting or galvanization). Finally, it is necessary to describe the external boundary conditions to the overpack situated on an ISFSI pad. An isolated HI-STORM will take suction of cool air from and reject heated air to, a semi-infinite half-space. In a rectilinear HI-STORM array, however, the unit situated in the center of the grid is evidently hydraulically most disadvantaged, because of potential interference to air intake from surrounding casks. To simulate this condition in a conservative manner, we erect a hypothetical cylindrical barrier around the centrally local HI-STORM. The radius of this hypothetical cylinder,  $R_o$ , is computed from the equivalent cask array downflow hydraulic diameter ( $D_h$ ) which is obtained as follows:

$$D_h = \frac{4 \times \text{Flow Area}}{\text{Wetted Perimeter}}$$

$$= \frac{4 \left( A_o - \frac{\pi}{4} d_o^2 \right)}{\pi d_o}$$

where:

- $A_o$  = Minimum tributary area ascribable to one HI-STORM (see Figure 4.4.24).
- $d_o$  = HI-STORM overpack outside diameter

The hypothetical cylinder radius,  $R_o$ , is obtained by adding half  $D_h$  to the radius of the HI-STORM overpack. In this manner, the hydraulic equivalence between the cask array and the HI-STORM overpack to hypothetical cylindrical annulus is established.

For purposes of the design basis analyses reported in this chapter, the tributary area  $A_o$  is assumed to be equal to 346 sq. ft. Sensitivity studies on the effect of the value of  $A_o$  on the thermal performance of the HI-STORM System shows that the system response is essentially insensitive to the assumed value of the tributary area. For example, a thermal calculation using  $A_o=225$  sq. ft. (corresponding to 15 ft. square pitch) and design basis heat load showed that the peak cladding temperature is less than 1°C greater than that computed using  $A_o=346$  sq. ft. Therefore, the distance between the vertically arrayed HI-STORMs in an ISFSI should be guided by the practical (rather than thermal) considerations, such as personnel access to maintain air ducts or painting the cask external surfaces.

The internal surface of the hypothetical cylinder of radius  $R_o$  surrounding the HI-STORM module is conservatively assumed to be insulated. Any thermal radiation heat transfer from the HI-STORM overpack to this insulated surface will be perfectly reflected, thereby bounding radiative blocking from neighboring casks. Then, in essence, the HI-STORM module is assumed to be confined in a large cylindrical “tank” whose wall surface boundaries are modeled as zero heat flux boundaries. The air in the “tank” is the source of “feed air” to the overpack. The air in the tank is replenished by ambient air from above the top of the HI-STORM overpacks. There are two sources of heat input to the exposed surface of the HI-STORM overpack. The most important source of heat input is the internal heat generation within the MPC. The second source of heat input is insolation, which is conservatively quantified in the manner of the preceding subsection.

The FLUENT model consisting of the axisymmetric 3-D MPC space, the overpack, and the enveloping tank is schematically illustrated in Figure 4.4.13. The HI-STORM thermosiphon-enabled solution is computed in a two-step process. In the first step, a HI-STORM overpack thermal model computes the ventilation effect from annulus heating by MPC decay heat. In this model, heat dissipation is conservatively restricted to the MPC shell (i.e., heat dissipation from MPC lid and baseplate completely neglected. This modeling assumption has the effect of overstating the MPC shell, annulus air and concrete temperatures. In the next step, the temperature of stored fuel in a pressurized helium canister (thermosiphon model) is determined using the overpack thermal solution in the first step to fashion a bounding MPC shell temperature profile for the MPC thermal model. The modeling details are provided in the Holtec benchmarking report [4.1.54.4-12]. A summary of the essential features of this model is presented in the following:

- A conservatively lower bound canister pressure of 5 atm is postulated for the thermosiphon modeling.
- Heat input due to insolation is applied to the top surface and the cylindrical surface of the overpack with a bounding maximum solar absorbtivity equal to 1.0.
- The heat generation in the MPC is assumed to be uniform in each horizontal plane, but to vary in the axial direction to correspond to the axial power distribution listed in Chapter 2.

- The most disadvantageously placed cask (i.e., the one subjected to maximum radiative blockage), is modeled.
- The bottom surface of the overpack, in contact with the ISFSI pad, rejects heat through the pad to the constant temperature (77°F) earth below. For some scenarios, the bottom surface of the overpack is conservatively assumed to be adiabatic.

The finite-volume model constructed in this manner will produce an axisymmetric temperature distribution. The peak temperature will occur at the centerline and is expected to be above the axial location of peak heat generation. As will be shown in Subsection 4.4.2, the results of the finite-volume solution bear out these observations.

The HI-STORM 100 System is evaluated for two fuel storage scenarios. In one scenario, designated as uniform loading, every basket cell is assumed to be occupied with fuel producing heat at the maximum rate. Storage of moderate burnup and high burnup fuels are analyzed for this loading scenario. In another scenario, denoted as regionalized loading, a two-region fuel loading configuration is stipulated. The two regions are defined as an inner region (for storing hot fuel) and an outer region with low decay heat fuel physically enveloping the inner region. This scenario is depicted in Figure 4.4.25. The inner region is shown populated with fuel having a heat load of  $q_1$  and post-core decay time (PCDT) or age  $\tau$ , and the outer region with fuel of heat load  $q_2$  and age  $\tau_2$ , where  $q_1 > q_2$ . For conservatism the outer region fuel permissible cladding temperature ( $T_2$ ) is assumed to be that of old fuel ( $\tau = 15$  years). By ensuring that the interface boundary temperature is less than or equal to  $T_2$  ensures that fuel in the outer region is below permissible temperatures for any fuel age. To permit hot fuel storage in the inner region, a uniform low decay heat rate is stipulated for the outer region fuel. The maximum allowable heat load for inner region fuel ( $q_1$ ), then, is a function of fuel age dependent permissible temperature set forth in Table 4.3.7 and Appendix 4.A for moderate and high burnup fuels, respectively. For the regionalized loading scenario, the most restrictive of the two burnups dependent permissible temperature limits is used in the thermal evaluation. In the HI-STORM 100 System, four central locations in the MPC-24 and MPC-24E, twelve inner cells in MPC-32 and 32 in MPC-68 are designated as inner region locations in the regionalized fuel-loading scenario. Results of thermal evaluations for both scenarios are present in Subsection 4.4.2.

#### 4.4.1.1.10 Effect of Fuel Cladding Crud Resistance

In this subsection, a conservatively bounding estimate of temperature drop across a crud film adhering to a fuel rod during dry storage conditions is determined. The evaluation is performed for a BWR fuel assembly based on an upper bound crud thickness obtained from the PNL-4835 report ([4.3.2], Table 3). The crud present on the fuel assemblies is predominately iron oxide mixed with small quantities of other metals such as cobalt, nickel, chromium, etc. Consequently, the effective conductivity of the crud mixture is expected to be in the range of typical metal alloys. Metals have thermal conductivities several orders of magnitude larger than that of helium. In the interest of extreme conservatism, however, a film of helium with the same thickness replaces the crud layer. The calculation is performed in two steps. In the first step, a crud film resistance is determined based

on a bounding maximum crud layer thickness replaced with a helium film on the fuel rod surfaces. This is followed by a peak local cladding heat flux calculation for the GE 7×7 array fuel assembly postulated to emit a conservatively bounding decay heat equal to 0.5kW. The temperature drop across the crud film obtained as a product of the heat flux and crud resistance terms is determined to be less than 0.1°F. The calculations are presented below.

Bounding Crud Thickness(s) = 130μm (4.26×10<sup>-4</sup> ft) (PNL-4835)  
 Crud Conductivity (K) = 0.1 Btu/ft-hr-°F (conservatively assumed as helium)  
 GE 7×7 Fuel Assembly:

Rod O.D. = 0.563"  
 Active Fuel Length = 150"  
 Heat Transfer Area = (7×7) × (π×0.563) × (150/144) = 90.3 ft<sup>2</sup>  
 Axial Peaking Factor = 1.195 (Burnup distribution Table 2.1.11)  
 Decay Heat = 500W (conservative assumption)

$$\text{Crud Resistance} = \frac{\delta}{K} = \frac{4.26 \times 10^{-4}}{0.1} = 4.26 \times 10^{-3} \frac{\text{ft}^2 \cdot \text{hr} \cdot ^\circ\text{F}}{\text{Btu}}$$

$$\begin{aligned} \text{Peak Heat Flux} &= \frac{(500 \times 3.417) \text{ Btu/hr}}{90.3 \text{ ft}^2} \times 1.195 \\ &= 18.92 \times 1.195 = 22.6 \frac{\text{Btu}}{\text{ft}^2 \text{ hr}} \end{aligned}$$

Temperature drop (ΔT<sub>c</sub>) across crud film

$$\begin{aligned} &= 4.26 \times 10^{-3} \frac{\text{ft}^2 \cdot \text{hr} \cdot ^\circ\text{F}}{\text{Btu}} \times 22.6 \frac{\text{Btu}}{\text{ft}^2 \cdot \text{hr}} \\ &= 0.096^\circ\text{F} \end{aligned}$$

(i. e., less than 0.1°F)

Therefore, it is concluded that deposition of crud does not materially change the SNF cladding temperature.

#### 4.4.1.1.11 Thermal Conductivity Calculations with Diluted Backfill Helium

In this subsection, the thermal conductivities of mixtures of the helium backfill gas and the gaseous fission products released from a hypothetical rupture of 10% of the stored fuel rods are evaluated. The gaseous fission products release fractions are stipulated in NUREG-1536. The released gases will mix with the helium backfill gas and reduce its thermal conductivity. These reduced thermal conductivities are applied to determine fuel assembly, and MPC fuel basket and basket periphery effective conductivities for thermal evaluation of the HI-STORM System.

Appendix C of NUREG/CR-0497 [4.4.7] describes a method for calculating the effective thermal conductivity of a mixture of gases. The same method is also described by Rohsenow and Hartnett [4.2.2]. The following expression is provided by both references:

$$k_{\text{mix}} = \sum_{i=1}^n \left( \frac{k_i x_i}{x_i + \sum_{\substack{j=1 \\ j \neq i}}^n \phi_{ij} x_j} \right)$$

where:

- $k_{\text{mix}}$  = thermal conductivity of the gas mixture (Btu/hr-ft-°F)
- $n$  = number of gases
- $k_i$  = thermal conductivity of gas component  $i$  (Btu/hr-ft-°F)
- $x_i$  = mole fraction of gas component  $i$

In the preceding equation, the term  $\phi_{ij}$  is given by the following:

$$\phi_{ij} = \phi_{ij} \left[ 1 + 2.41 \frac{(M_i - M_j)(M_i - 0.142 \cdot M_j)}{(M_i + M_j)^2} \right]$$

where  $M_i$  and  $M_j$  are the molecular weights of gas components  $i$  and  $j$ , and  $\phi_{ij}$  is:

$$\phi_{ij} = \frac{\left[ 1 + \left( \frac{k_i}{k_j} \right)^{\frac{1}{2}} \left( \frac{M_i}{M_j} \right)^{\frac{1}{4}} \right]^2}{2^{\frac{3}{2}} \left( 1 + \frac{M_i}{M_j} \right)^{\frac{1}{2}}}$$

Table 4.4.7 presents a summary of the gas mixture thermal conductivity calculations for the MPC-24 and MPC-68 MPC designs containing design basis fuel assemblies.

Having calculated the gas mixture thermal conductivities, the effective thermal conductivities of the design basis fuel assemblies are calculated using the finite-volume model described in Subsection 4.4.1.1.2. Only the helium gas conductivity is changed, all other modeling assumptions are the same. The fuel assembly effective thermal conductivities with diluted helium are compared to those with undiluted helium in Table 4.4.8. From this table, it is observed that a 10% rod rupture condition has a relatively minor impact on the fuel assembly effective conductivity. Because the fuel regions comprise only a portion of the overall fuel basket thermal conductivity, the 10% rod rupture condition will have an even smaller impact on the basket effective conductivity.

#### 4.4.1.1.12 Effects of Hypothetical Low Fuel Rod Emissivity

The value of emissivity ( $\epsilon$ ) utilized in this FSAR was selected as 0.8 based on:

- i. the recommendation of an EPRI report [4.1.3]

- ii. Holtec’s prior licensing experience with the HI-STAR 100 System
- iii. other vendors’ cask licensing experience with the NRC
- iv. authoritative literature citations

The table below provides relevant third party information to support the emissivity value utilized in this FSAR.

<b>Source</b>	<b>Reference</b>	<b>Zircaloy Emissivity</b>
EPRI	[4.1.3]	0.8
TN-68 TSAR	Docket 72-1027	0.8
TN-40	Prairie Island Site Specific ISFSI	0.8
TN-32	Docket 72-1021	0.8
Todreas & Mantuefel	[4.4.8]	0.8
DOE SNF Report	[4.4.9]	0.8

The appropriateness of the selected value of  $\epsilon$  is further supported by the information provided by PNL-4835 [4.3.2] and NUREG/CR-0497 [4.4.7]. PNL-4835 reports cladding oxidation thickness in U.S. Zircaloy LWR SNF assemblies (20  $\mu\text{m}$  for PWR and 30  $\mu\text{m}$  for BWR fuel). If these oxide thickness values are applied to the mathematical formulas presented for emissivity determination in [4.4.7], then the computed values are slightly higher than our assumed value of 0.8. It should be recognized that the formulas in [4.4.7] include a conservative assumption that depresses the value of computed emissivity, namely, absence of crud. Significant crud layers develop on fuel cladding surfaces during in-core operation. Crud, which is recognized by the above-mentioned NUREG document as having a boosting effect on  $\epsilon$ , is completely neglected.

The above discussion provides a reasonable rationale for our selection of 0.8 as the value for  $\epsilon$ . However, to determine the effect of a hypothetical low emissivity of 0.4, an additional thermal analysis adopting this value has been performed. In this analysis, each fuel rod of a fuel assembly is stipulated to have this uniformly low  $\epsilon = 0.4$  and the effective fuel thermal conductivity is recalculated. In the next step, all cells of an MPC basket are assumed to be populated with this low  $\epsilon$  fuel that is further assumed to be emitting decay heat at design basis level. The effective conductivity of this basket populated with low  $\epsilon$  fuel is recalculated. Using the recalculated fuel basket conductivity, the HI-STORM system temperature field is recomputed. This exercise is performed for the MPC-24 basket because, as explained in the next paragraph, this basket design, which accommodates a fewer number of fuel assemblies (compared to the MPC-68 and MPC-32) has a higher sensitivity to the emissivity parameter. This analysis has determined that the impact of a low  $\epsilon$  assumption on the peak cladding temperature is quite small (about 5°C). It is noted that these sensitivity calculations were performed under the completely suppressed helium thermosiphon cooling assumption. Consequently, as the burden of heat dissipation shouldered by radiation heat



transfer under this assumption is much greater, the resultant computed sensitivity is a conservative upper bound for the HI-STORM system.

The relatively insignificant increase in the computed peak clad temperature as a result of applying a large penalty in  $\epsilon$  (50%) is consistent with the findings in a German Ph.D. dissertation [4.4.11]. Dr. Anton's study consisted of analyzing a cask containing 4 fuel assemblies with a total heat load of 17 kW and helium inside the fuel cavity. For an emissivity of 0.8, the calculated peak cladding temperature was 337°C. In a sensitivity study, wherein the emissivity was varied from 0.7 to 0.9, the temperature changed only by 5°C, i.e. to 342°C and 332°C. Dr. Anton ascribed two reasons for this low impact of emissivity on computed temperatures. Although the radiative heat emission by a surface decreases with lower emissivity, the fraction of heat reflected from other surfaces increases. In other words, the through-assembly heat dissipation by this means increases thereby providing some compensation for the reduced emission. Additionally, the fourth power of temperature dependence of thermal radiation heat transfer reduces the impact of changes in the coefficients on computed temperatures. For storage containers with larger number of fuel assemblies (like the HI-STORM System), an even smaller impact would be expected, since a larger fraction of the heat is dissipated via the basket conduction heat transfer.

#### 4.4.1.1.13 HI-STORM Temperature Field with Low Heat Emitting Fuel

The HI-STORM 100 thermal evaluations for BWR fuel are grouped in two categories of fuel assemblies proposed for storage in the MPC-68. The two groups are classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE group of fuel assemblies are characterized by low burnup, long cooling time, and short active fuel lengths. Consequently, their heat loads are dwarfed by the DB group of fuel assemblies. The Dresden-1 (6x6 and 8x8), Quad+, and Humboldt Bay (7x7 and 6x6) fuel assemblies are grouped as the LHE fuel. This fuel is evaluated when encased in Damaged Fuel Containers (DFC). As a result of interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary, this configuration is bounding for thermal evaluation. In Table 4.4.2, two canister types for encasing LHE fuel are evaluated – a Holtec design and an existing canister in which some of the Dresden-1 fuel is currently stored (Transnuclear D-1 canister). The most resistive LHE fuel assembly (Dresden-1 8x8) is considered for thermal evaluation (see Table 4.4.2) in a DFC container. The MPC-68 basket effective conductivity, loaded with the most resistive fuel assembly (encased in a canister) is provided in Table 4.4.3. To this basket, LHE decay heat is applied and a HI-STORM 100 System thermal solution computed. The peak cladding temperature is computed as 513°F, which is substantially below the temperature limit for long cooled fuel (~635/752°F).

A thorium rod canister designed for holding a maximum of twenty fuel rods arrayed in a 5x4 configuration is currently stored at the Dresden-1 spent fuel pool. The fuel rods were originally constituted as part of an 8x8 fuel assembly and used in the second and third cycle of Dresden-1 operation. The maximum fuel burnup of these rods is quite low (~14,400 MWD/MTU). The thorium rod canister internal design is a honeycomb structure formed from 12-gage stainless steel plates. The rods are loaded in individual square cells. This long cooled, part assembly (18 fuel rods) and very low fuel burnup thorium rod canister renders it a miniscule source of decay heat. The canister all-metal internal honeycomb construction serves as an additional means of heat dissipation in the fuel cell space. In accordance with fuel loading stipulation in the Technical Specifications, long cooled fuel is

loaded toward the basket periphery (i.e., away from the hot central core of the fuel basket). All these considerations provide ample assurance that these fuel rods will be stored in a benign thermal environment and, therefore, remain protected during long-term storage.

#### 4.4.1.2 Test Model

A detailed analytical model for thermal design of the HI-STORM System was developed using the FLUENT CFD code and the industry standard ANSYS modeling package, as discussed in Subsection 4.4.1.1. As discussed throughout this chapter and specifically in Section 4.4.6, the analysis incorporates significant conservatism so as to compute bounding fuel cladding temperatures. Furthermore, compliance with specified limits of operation is demonstrated with adequate margins. In view of these considerations, the HI-STORM System thermal design complies with the thermal criteria set forth in the design basis (Sections 2.1 and 2.2) for long-term storage under normal conditions. Additional experimental verification of the thermal design is therefore not required.

#### 4.4.2 Maximum Temperatures

All four MPC-basket designs developed for the HI-STORM System have been analyzed to determine temperature distributions under long-term normal storage conditions, and the results summarized in this subsection. A cross-reference of HI-STORM thermal analyses at other conditions with associated subsection of the FSAR summarizing obtained results is provided in Table 4.4.22. The MPC baskets are considered to be fully loaded with design basis PWR or BWR fuel assemblies, as appropriate. The systems are arranged in an ISFSI array and subjected to design basis normal ambient conditions with insolation.

As discussed in Subsection 4.4.1.1.1, the thermal analysis is performed using a submodeling process where the results of an analysis on an individual component are incorporated into the analysis of a larger set of components. Specifically, the submodeling process yields directly computed fuel temperatures from which fuel basket temperatures are then calculated. This modeling process differs from previous analytical approaches wherein the basket temperatures were evaluated first and then a basket-to-cladding temperature difference calculation by Wooten-Epstein or other means provided a basis for cladding temperatures. Subsection 4.4.1.1.2 describes the calculation of an effective fuel assembly thermal conductivity for an equivalent homogenous region. It is important to note that the result of this analysis is a function of thermal conductivity versus temperature. This function for fuel thermal conductivity is then input to the fuel basket effective thermal conductivity calculation described in Subsection 4.4.1.1.4. This calculation uses a finite-element methodology, wherein each fuel cell region containing multiple finite-elements has temperature-varying thermal conductivity properties. The resultant temperature-varying fuel basket thermal conductivity computed by this basket-fuel composite model is then input to the fuel basket region of the FLUENT cask model.

Because the FLUENT cask model incorporates the results of the fuel basket submodel, which in turn incorporates the fuel assembly submodel, the peak temperature reported from the FLUENT model is the peak temperature in any component. In a dry storage cask, the hottest components are the fuel assemblies. It should be noted that, because the fuel assembly models described in Subsection 4.4.1.1.2 include the fuel pellets, the FLUENT calculated peak temperatures reported in Tables 4.4.9

and 4.4.10 are actually peak pellet centerline temperatures which bound the peak cladding temperatures, and are therefore conservatively reported as the cladding temperatures.

Applying the radiative blocking factor applicable for the worst case cask location, conservatively bounding axial temperatures at the most heated fuel cladding are shown in Figures 4.4.16 and 4.4.17 for MPC-24 and MPC-68 to depict the thermosiphon effect in PWR and BWR SNF. From these plots, the upward movement of the hot spot is quite evident. As discussed in this chapter, these calculated temperature distributions incorporate many conservatisms. The maximum fuel clad temperatures for zircaloy clad fuel assemblies are listed in Tables 4.4.9, 4.4.10, 4.4.26, and 4.4.27, which also summarize maximum calculated temperatures in different parts of the MPCs and HI-STORM overpack (Table 4.4.36).

Figures 4.4.19 and 4.4.20, respectively, depict radial temperature distribution in the PWR (MPC-24) and the BWR (MPC-68) at the horizontal plane where maximum fuel cladding temperature occurs. Finally, axial variations of the ventilation air temperatures and that of the inner shell surface are depicted in Figure 4.4.26 for a bounding heat load.

The following additional observations can be derived by inspecting the temperature field obtained from the finite volume analysis:

- The fuel cladding temperatures *are below the regulatory limit (ISG-11 [4.1.4]) under all storage scenarios (uniform and regionalized) in all MPCs* ~~are in compliance with the temperature limits determined using both the DCCG methodology [4.3.5] and the PNL CSFM methodology [4.3.1].~~
- The maximum temperature of the basket structural material is within the stipulated design temperature.
- The maximum temperature of the ~~Boral~~ neutron absorber is below the *design temperature* ~~material supplier's recommended limit~~.
- The maximum temperatures of the MPC pressure boundary materials are well below their respective ASME Code limits.
- The maximum temperatures of concrete are within the *guidance of the governing ACI Code* ~~NRC's recommended limits [4.4.10] (See Table 4.3.1-)~~.

~~Noting that the permissible peak cladding temperature is a function of fuel age, parametric peak fuel cladding temperature versus total decay heat load information is computed from the FLUENT thermal model solution. The allowable fuel cladding temperature limits are presented in Section 4.3 for moderate burnup fuel and in Appendix 4.A for high burnup fuel.~~

~~Because the peak clad temperature limits are dependent on burnup and the fuel age at the start of dry storage, the allowable decay heat load is also dependent on these parameters. Tables 4.4.20, , 4.4.21, 4.4.28, and 4.4.29, for the MPC-24 and MPC-68, MPC-32 and MPC-24E, respectively, present the~~

allowable decay heat load as a function of fuel age for moderate burnup fuel. Tables 4.4.32 through 4.4.35 present the results for high burnup fuel. Burnup and cooling time curves, developed in source-term calculations in Chapter 5 and reported in Chapter 2, are generated from the heat load limits in those tables. It is noted that the burnup and cooling time curves are developed for the most limiting fuel assembly<sup>†</sup> of each type (PWR and BWR), but are applied to all assemblies of each type. By definition, the limiting fuel assembly emits more heat than any other assembly of its type at a given burnup and cooling time does. Thus, if the limiting fuel assembly meets the allowable clad temperature limit by a certain margin, then the other fuel assemblies of its type with equal burnup and cooling time will meet the clad temperature limit by an even greater margin. The added margin can be quite considerable. For example, the design-basis PWR assembly is the B&W 15×15, which is used to determine Technical Specification limits for burnup in the HI-STORM System. For certain Westinghouse fuel types, the decay heat loads corresponding to these burnup limits will be about 15% less than that of the design-basis assembly. This decay heat over-prediction for other than design-basis assemblies renders the predicted peak temperatures extremely conservative for those assemblies.

For the regionalized loading scenario as depicted in Figure 4.4.25, outer region decay heat limits are stipulated in Table 4.4.30. The inner region heat load limit will be governed by the peak cladding temperature limit for the hot fuel, provided that the interface cladding temperature limit for long cooled fuel is not exceeded. The MPC-32 and MPC-68 heat load limits are determined by analysis to be governed by this requirement. In the MPC-24 and MPC-24E regionalized loading scenarios, the interface cladding temperature limit is reached first for certain fuel cooling times. Thus, the peak cladding temperatures for these MPCs are below their permissible values by a greater margin. The inner region heat load limits are provided in Table 4.4.31.

The calculated temperatures are based on a series of analyses, described previously in this chapter, that incorporate many conservatisms. A list of the significant conservatisms is provided in Subsection 4.4.6. As such, the calculated temperatures are upper bound values that would exceed actual temperatures.

The above observations lead us to conclude that the temperature field in the HI-STORM System with a fully loaded MPC containing design-basis heat emitting SNF complies with all regulatory and industry temperature limits. In other words, the thermal environment in the HI-STORM System will be conducive to long-term safe storage of spent nuclear fuel.

#### 4.4.3 Minimum Temperatures

In Table 2.2.2 of this report, the minimum ambient temperature condition for the HI-STORM storage overpack and MPC is specified to be -40°F. If, conservatively, a zero decay heat load with no solar input is applied to the stored fuel assemblies, then every component of the system at steady state would be at a temperature of -40°F. All HI-STORM storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode at this minimum

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<sup>†</sup> The limiting fuel assembly (also referred to as the design-basis assembly) is defined as that assembly which is the most heat emissive of its type (PWR or BWR) as a given burnup and cooling time.

temperature condition. Structural evaluations in Chapter 3 show the acceptable performance of the overpack and MPC steel and concrete materials at low service temperatures. Criticality and shielding evaluations (Chapters 5 and 6) are unaffected by temperature.

#### 4.4.4 Maximum Internal Pressure

The MPC is initially filled with dry helium after fuel loading and drying prior to installing the MPC closure ring. During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature as determined based on the thermal analysis methodology described earlier. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined based on the ideal gas law, which states that the absolute pressure of a fixed volume of gas is proportional to its absolute temperature. Tables 4.4.12, 4.4.13, 4.4.24, and 4.4.25 present summaries of the calculations performed to determine the net free volume in the MPC-24, MPC-68, MPC-32, and MPC-24E, respectively.

The MPC maximum gas pressure is considered for a postulated accidental release of fission product gases caused by fuel rod rupture. For these fuel rod rupture conditions, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the Ideal Gas Law. Based on fission gases release fractions (per NUREG 1536 criteria [4.4.10]), net free volume and initial fill gas pressure, the bounding maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in Table 4.4.14. The maximum gas pressures listed in Table 4.4.14 are all below the MPC internal design pressure listed in Table 2.2.1.

The inclusion of PWR non-fuel hardware (BPRAs control elements and thimble plugs) to the PWR baskets influences the MPC internal pressure through two distinct effects. The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures as well as the temperature of the gas filling the space between fuel rods. The gas volume displaced by the mass of non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influence on the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STORM thermal analysis, the computed temperature field (with non-fuel hardware excluded) has been determined to provide a conservatively bounding temperature field for the PWR baskets (MPC-24, MPC-24E, and MPC-32). The MPC cavity free space is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included. *This approach ensures conservative bounding pressures.*

During in-core irradiation of BPRAs, neutron capture by the B-10 isotope in the neutron absorbing material produces helium. Two different forms of the neutron absorbing material are used in BPRAs: Borosilicate glass and B<sub>4</sub>C in a refractory solid matrix (Al<sub>2</sub>O<sub>3</sub>). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin-walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operation conditions (2,300 psia and approximately 600°F coolant temperature). The B<sub>4</sub>C- Al<sub>2</sub>O<sub>3</sub> neutron

absorber material is principally used in B&W and CE fuel BPRA designs. The relatively low temperature of the poison material in BPRA rods (relative to fuel pellets) favor the entrapment of helium atoms in the solid matrix.

Several BPRA designs are used in PWR fuel that differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and W-15x15) has used 6, 12, 16, and 20 rods per assembly BPRAs and the later (W-17x17) fuel uses up to 24 rods per BPRA. The BPRA rods in the older fuel are much larger than the later fuel and, therefore, the B-10 isotope inventory in the 20-rod BPRAs bounds the newer W-17x17 fuel. Based on bounding BPRA rods internal pressure, a large hypothetical quantity of helium (7.2 g-moles/BPRA) is assumed to be available for release into the MPC cavity from each fuel assembly in the PWR baskets. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.14.

#### 4.4.5 Maximum Thermal Stresses

~~Thermal expansion induced mechanical stresses due to non-uniform temperature distributions are reported in Chapter 3 of this report. Table 4.4.15 provides a summary of HI-STORM System component temperature inputs for structural evaluation. Table 4.4.19 provides a summary of confinement boundary temperatures during normal storage conditions. Structural evaluation in Section 3.4.4 references these temperature results to demonstrate confinement boundary integrity. Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal stresses in load bearing members, the HI-STORM System is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansion calculations are performed to demonstrate that engineered gaps in the HI-STORM System are adequate to accommodate thermal expansion. To facilitate structural integrity evaluations, temperature distributions are provided herein (Tables 4.4.9, 4.4.10, 4.4.26 and 4.4.27).~~

*As stated above, the HI-STORM System is engineered with gaps for the fuel basket and MPC to thermally expand without restraint of free end expansion. Differential thermal expansion of the following gaps are evaluated:*

- a. Fuel Basket-to-MPC Radial Gap*
- b. Fuel Basket to MPC Axial Gap*
- c. MPC-to-Overpack Radial Gap*
- d. MPC-to-Overpack Axial Gap*

*To demonstrate that the fuel basket and MPC are free to expand without restraint, it is required to show that differential thermal expansion from fuel heatup is less than the as-built gaps that exist in the HI-STORM System. For this purpose a suitably bounding temperature profile ( $T(r)$ ) for the fuel basket is established in Figure 4.4.27 wherein the center temperature ( $TC$ ) is set at the limit (752 °F) for fuel cladding (conservatively bounding assumption) and the basket periphery ( $TP$ ) conservatively postulated at an upperbound of 600 °F (see Tables 4.4.9, 4.4.10, 4.4.26 and 4.4.27 for the maximum computed basket periphery temperatures). To maximize the fuel basket differential thermal expansion, the basket periphery-to-MPC shell temperature difference is conservatively maximized*

( $\Delta T = 175^\circ F$ ). From the bounding temperature profile  $T(r)$  and  $\Delta T$ , the mean fuel basket temperature ( $T1$ ) and MPC shell temperature ( $T2$ ) are computed as follows:

$$T1 = \frac{\int_0^1 rT(r)dr}{\int_0^1 r dr} = 676^\circ F$$

$$T2 = TP - \Delta T = 425^\circ F$$

The differential radial growth of the fuel basket ( $Y1$ ) from an initial reference temperature ( $To = 70^\circ F$ ) is computed as:

$$Y1 = R \times [A1 \times (T1 - To) - A2 \times (T2 - To)]$$

where:

$R$  = Basket radius (conservatively assumed to be the MPC radius)

$A1, A2$  = Coefficients of thermal expansion for fuel basket and MPC shell at  $T1$  and  $T2$  respectively for Alloy X (Chapter 1 and Table 3.3.1)

For computing the relative axial growth of the fuel basket in the MPC, bounding temperatures for the fuel basket ( $TC$ ) and MPC shell temperature  $T2$  computed above (assuming a maximum basket periphery-to-MPC shell temperature differential) are adopted. The differential expansion is computed by a formula similar to the one for radial growth after replacing  $R$  with basket height ( $H$ ), which is conservatively assumed to be that of the MPC cavity.

For computing the radial and axial MPC-to-overpack differential expansions, the MPC shell is postulated at its design temperature (Chapter 2, Table 2.2.3) and thermal expansion of the overpack is ignored. Even with the conservative computation of the differential expansions in the manner of the foregoing, it is evident from the data compiled in Table 4.4.37 that the differential expansions are a fraction of their respective gaps.

#### 4.4.6 Evaluation of System Performance for Normal Conditions of Storage

The HI-STORM System thermal analysis is based on a detailed and complete heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and overpack. The thermal model incorporates many conservative features that render the results for long-term storage to be extremely conservative:

1. The most severe levels of environmental factors for long-term normal storage, which are an ambient temperature of  $80^\circ F$  and 10CFR71 insulation levels, were coincidentally imposed on the system.
2. A hypothetical rupture of 10% of the stored fuel rods was conservatively considered for determining the thermal conductivity of the diluted helium backfill gas.

3. The most adversely located\* HI-STORM System in an ISFSI array was considered for analysis.
4. A conservative assessment of thermosiphon effect in the MPC, which is intrinsic to the HI-STORM fuel basket design is included in the thermal analyses.
5. *The MPC internal pressure is conservatively understated for performing temperature calculations. This maximizes calculated temperatures.* ~~Not Used~~
6. No credit was considered for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports. The fuel assemblies and MPC basket were conservatively considered to be in concentric alignment.
7. The MPC is assumed to be loaded with the SNF type which has the maximum equivalent thermal resistance of all fuel types in its category (BWR or PWR), as applicable.
8. The design basis maximum decay heat loads are used for all thermal-hydraulic analyses. For casks loaded with fuel assemblies having decay heat generation rates less than design basis, additional thermal margins of safety will exist. This is assured by defining the burnup limits, ~~as a function of age,~~ for the fuel assemblies based on the bounding (i.e., most heat emissive) fuel assembly types within each class (PWR or BWR). ~~As demonstrated in the source term calculations described Chapter 5, the B&W 15×15 and GE 7×7 are the governing PWR and BWR fuel assemblies, respectively.~~ For all other fuel types, the heat emission rates at the design-basis burnup levels will be below the design-basis heat emission rate.
9. Not Used
- ~~10. The enhancement of heat transfer owing to the so-called “Rayleigh effect” in the basket/MPC interface region, which was included in the analyses underlying the original CoC on the HI-STORM 100 System, is neglected in this revision of the SAR for conservatism.~~
- ~~11. Aluminum heat conduction elements ignored in the thermal analyses.~~

Temperature distribution results obtained from this highly conservative thermal model show that the maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be much greater due to the many conservative assumptions incorporated in the analysis. The long-term impact of decay heat induced temperature levels on the HI-STORM System structural and neutron shielding materials is considered to be negligible. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep-induced degradation. Furthermore, stresses induced due

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\* *In an ISFSI array, HI-STORM overpacks at interior locations are relatively more disadvantaged in their lateral access to ambient air and in their effectiveness to radiate heat to the environment. To bound these effects, a reference cask is enclosed in a hypothetical reflecting cylinder as described in Section 4.4.1.1.9.*



to imposed temperature gradients are within Code limits. Therefore, it is concluded that the HI-STORM System thermal design is in compliance with 10CFR72 requirements.

Table 4.4.1

SUMMARY OF PWR FUEL ASSEMBLY EFFECTIVE  
THERMAL CONDUCTIVITIES

<b>Fuel</b>	<b>@ 200°F (Btu/ft-hr-°F)</b>	<b>@ 450°F (Btu/ft-hr-°F)</b>	<b>@ 700°F (Btu/ft-hr-°F)</b>
W - 17×17 OFA	0.182	0.277	0.402
W - 17×17 Standard	0.189	0.286	0.413
W - 17×17 Vantage	0.182	0.277	0.402
W - 15×15 Standard	0.191	0.294	0.430
W - 14×14 Standard	0.182	0.284	0.424
W - 14×14 OFA	0.175	0.275	0.413
B&W - 17×17	0.191	0.289	0.416
B&W - 15×15	0.195	0.298	0.436
CE - 16×16	0.183	0.281	0.411
CE - 14×14	0.189	0.293	0.435
HN <sup>†</sup> - 15×15 SS	0.180	0.265	0.370
W - 14×14 SS	0.170	0.254	0.361
B&W-15x15 Mark B-11	0.187	0.289	0.424
CE-14x14 (MP2)	0.188	0.293	0.434
IP-1 (14x14) SS	0.125	0.197	0.293

<sup>†</sup> Haddam Neck Plant B&W or Westinghouse stainless steel clad fuel assemblies.

Table 4.4.2

SUMMARY OF BWR FUEL ASSEMBLY EFFECTIVE  
THERMAL CONDUCTIVITIES

<b>Fuel</b>	<b>@ 200°F (Btu/ft-hr-°F)</b>	<b>@ 450°F (Btu/ft-hr-°F)</b>	<b>@ 700°F (Btu/ft-hr-°F)</b>
Dresden 1 - 8×8 <sup>†</sup>	0.119	0.201	0.319
Dresden 1 - 6×6 <sup>†</sup>	0.126	0.215	0.345
GE - 7×7	0.171	0.286	0.449
GE - 7×7R	0.171	0.286	0.449
GE - 8×8	0.168	0.278	0.433
GE - 8×8R	0.166	0.275	0.430
GE10 - 8×8	0.168	0.280	0.437
GE11 - 9×9	0.167	0.273	0.422
AC <sup>††</sup> -10×10 SS	0.152	0.222	0.309
Exxon-10×10 SS	0.151	0.221	0.308
Damaged Dresden-1 8×8 <sup>†</sup> (in a Holtec damaged fuel container)	0.107	0.169	0.254
Humboldt Bay-7x7 <sup>†</sup>	0.127	0.215	0.343
Dresden-1 Thin Clad 6x6 <sup>†</sup>	0.124	0.212	0.343
Damaged Dresden-1 8x8 (in TN D-1 canister) <sup>†</sup>	0.107	0.168	0.252
8x8 Quad <sup>+</sup> Westinghouse <sup>†</sup>	0.164	0.276	0.435

<sup>†</sup> Cladding temperatures of low heat emitting Dresden (intact and damaged) SNF in the HI-STORM System will be bounded by design basis fuel cladding temperatures. Therefore, these fuel assembly types are excluded from the list of fuel assemblies (zircaloy clad) evaluated to determine the most resistive SNF type.

<sup>††</sup> Allis-Chalmers stainless steel clad fuel assemblies.

Table 4.4.3

MPC BASKET EQUIVALENT ISOTROPIC THERMAL CONDUCTIVITY VALUES<sup>††</sup>

<b>Basket</b>	<b>@200°F (Btu/ft-hr-°F)</b>	<b>@450°F (Btu/ft-hr-°F)</b>	<b>@700°F (Btu/ft-hr-°F)</b>
MPC-24 (Zircaloy Clad Fuel)	1.109	1.495	1.955
MPC-68 (Zircaloy Clad Fuel)	1.111	1.347	1.591
MPC-24 (Stainless Steel Clad Fuel) †	0.897	1.213	1.577(a)
MPC-68 (Stainless Steel Clad Fuel)†	1.070	1.270	1.451(b)
MPC-32 (Zircaloy Clad Fuel)	1.015	1.271	1.546
MPC-32 (Stainless Steel Clad Fuel)†	0.806	0.987	1.161 (c)
MPC-24E (Zircaloy Clad Fuel)	1.216	1.637	2.133
MPC-24E (Stainless Steel Clad fuel)†	0.991	1.351	1.766 (d)

- (a) Conductivity is 19% less than corresponding zircaloy fueled basket.
- (b) Conductivity is 9% less than corresponding zircaloy fueled basket.
- (c) Conductivity is 25% less than corresponding zircaloy fueled basket.
- (d) Conductivity is 17% less than corresponding zircaloy fueled basket.

<sup>††</sup> The values reported in this table are conservatively understated.

<sup>†</sup> Evaluated in a damaged fuel canister (conservatively bounding)

Table 4.4.4

~~[INTENTIONALLY DELETED] CLOSED-CAVITY NUSSELT NUMBER RESULTS  
FOR HELIUM FILLED MPC PERIPHERAL VOIDS<sup>†</sup>~~

<b>Temperature (°F)</b>	<b>Nusselt Number (PWR Baskets)</b>	<b>Nusselt Number (BWR Basket)</b>
200	3.17	2.41
450	2.56	1.95
700	2.21	1.68

<sup>†</sup> For conservatism the Rayleigh effect is ignored in the MPC thermal analyses.

Table 4.4.5

SUMMARY OF 10×10 ARRAY TYPE BWR FUEL ASSEMBLY  
EFFECTIVE THERMAL CONDUCTIVITIES<sup>†</sup>

<b>Fuel Assembly</b>	<b>@ 200°F (Btu/ft-hr-°F)</b>	<b>@ 450°F (Btu/ft-hr-°F)</b>	<b>@ 700°F (Btu/ft-hr-°F)</b>
GE-12/14	0.166	0.269	0.412
Atrium-10	0.164	0.266	0.409
SVEA-96	0.164	0.269	0.416

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<sup>†</sup> The conductivities reported in this table are obtained by the simplified method described in the beginning of Subsection 4.4.1.1.2.

Table 4.4.6

COMPARISON OF ATRIUM-10 BWR FUEL ASSEMBLY CONDUCTIVITY<sup>†</sup> WITH  
THE BOUNDING<sup>††</sup> BWR FUEL ASSEMBLY CONDUCTIVITY

Temperature (°F)	Atrium-10 BWR Assembly		Bounding BWR Assembly	
	(Btu/ft-hr-°F)	(W/m-K)	(Btu/ft-hr-°F)	(W/m-K)
200	0.225	0.389	0.171	0.296
450	0.345	0.597	0.271	0.469
700	0.504	0.872	0.410	0.710

<sup>†</sup> The reported effective conductivity has been obtained from a rigorous finite-element model.

<sup>††</sup> The bounding BWR fuel assembly conductivity applied in the MPC-68 basket thermal analysis.

Table 4.4.7

SUMMARY OF THERMAL CONDUCTIVITY CALCULATIONS  
FOR MPC HELIUM DILUTED BY RELEASED ROD GASES

Component Gas	Molecular Weight (g/mole)	Component Gas Mole Fractions and Mixture Conductivity (Btu/hr-ft-°F)	
		MPC-24	MPC-68
MPC Backfill Helium	4	0.951	0.962
Fuel Rod Backfill Helium	4	0.023	$5.750 \times 10^{-3}$
Rod Tritium	3	$1.154 \times 10^{-5}$	$4.483 \times 10^{-5}$
Rod Krypton	85	$2.372 \times 10^{-3}$	$2.905 \times 10^{-3}$
Rod Xenon	131	0.024	0.030
Rod Iodine	129	$1.019 \times 10^{-3}$	$1.273 \times 10^{-3}$
Mixture of Gases (diluted helium)	N/A	0.088 at 200°F 0.116 at 450°F 0.142 at 700°F	0.086 at 200°F 0.113 at 450°F 0.139 at 700°F



Table 4.4.8

COMPARISON OF COMPONENT THERMAL CONDUCTIVITIES  
WITH AND WITHOUT BACKFILL HELIUM DILUTION

	<b>@ 200°F</b> <b>(Btu/hr-ft-°F)</b>	<b>@ 450°F</b> <b>(Btu/hr-ft-°F)</b>	<b>@ 700°F</b> <b>(Btu/hr-ft-°F)</b>
GE-11 9×9 Fuel Assembly with Undiluted Helium	0.171	0.271	0.410
GE-11 9×9 Fuel Assembly with Diluted Helium	0.158	0.254	0.385
<u>W</u> 17×17 OFA Fuel Assembly with Undiluted Helium	0.257	0.406	0.604
<u>W</u> 17×17 OFA Fuel Assembly with Diluted Helium	0.213	0.347	0.537

Table 4.4.9

HI-STORM<sup>†</sup> SYSTEM LONG-TERM NORMAL  
STORAGE MAXIMUM TEMPERATURES  
(MPC-24 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	691	752787 <sup>††</sup>
MPC Basket	650	725 <sup>†††</sup>
Basket Periphery	486	725 <sup>†††</sup>
MPC Outer Shell	344	450

<sup>†</sup> Bounding overpack temperatures are provided in Table 4.4.36.

<sup>††</sup> ~~This temperature limit is in accordance with ISG-11 [4.1.4]. The temperature limit is in accordance with DCCG (gross rupture) criteria. Permissible peak cladding temperature is 691°F (PNL Criteria).~~

<sup>†††</sup> The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism to the analysis of the HI-STORM System.

Table 4.4.10

HI-STORM<sup>†</sup> SYSTEM LONG-TERM NORMAL  
STORAGE MAXIMUM TEMPERATURES  
(MPC-68 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	740	752 <sup>††</sup> 824 <sup>††</sup>
MPC Basket	720	725 <sup>†††</sup>
Basket Periphery	501	725 <sup>†††</sup>
MPC Outer Shell	347	450

<sup>†</sup> Bounding overpack temperatures are provided in Table 4.4.36.

<sup>††</sup> ~~This temperature limit is in accordance with ISG-11 [4.1.4]. The temperature limit is in accordance with DCCG (gross rupture) criteria. Permissible cladding temperature is 742°F (PNL criteria).~~

<sup>†††</sup> The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism to the analysis of the HI-STORM System.

Table 4.4.11

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Table 4.4.12

SUMMARY OF MPC-24 FREE VOLUME CALCULATIONS

Item	Volume (ft <sup>3</sup> )
Cavity Volume	367.9
Basket Metal Volume	39.7
Bounding Fuel Assemblies Volume	78.8
Basket Supports and Fuel Spacers Volume	6.1
<del>Aluminum Conduction Elements</del>	<del>5.9<sup>†</sup></del>
Net Free Volume	<del>237.5 (6,724 liters)</del> 243.3 (6,889 liters)*

<sup>†</sup> ~~Bounding 1,000 lbs weight assumed.~~

\* *A conservative lowerbound value of 237.5 ft<sup>3</sup> (6,724 liters) is used for subsequent MPC internal pressure calculations.*

Table 4.4.13

SUMMARY OF MPC-68 FREE VOLUME CALCULATIONS

Item	Volume (ft <sup>3</sup> )
Cavity Volume	367.3
Basket Metal Volume	34.8
Bounding Fuel Assemblies Volume	93.0
Basket Supports and Fuel Spacers Volume	11.3
Aluminum Conduction Elements	5.9 <sup>†</sup>
Net Free Volume	222.3 (6,294 liters)

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<sup>†</sup> Bounding 1,000 lbs weight assumed. *Included herein to bound early production units with these optional items installed.*

Table 4.4.14  
SUMMARY OF MPC CONFINEMENT BOUNDARY PRESSURES<sup>†</sup>  
FOR LONG-TERM STORAGE

Condition	Pressure (psig)
MPC-24:	
Initial backfill (at 70°F)	31.3
Normal condition	66.4
With 1% rods rupture	66.1
With 10% rods rupture	72.2
With 100% rods rupture	132.5
MPC-68:	
Initial backfill (at 70°F)	31.3
Normal condition	67.1
With 1% rods rupture	67.5
With 10% rods rupture	71.1
With 100% rods rupture	107.6
MPC-32:	
Initial backfill (at 70°F)	31.3
Normal Condition	65.6
With 1% rods rupture	66.5
With 10% rods rupture	75.0
With 100% rods rupture	160.1
MPC-24E:	
Initial backfill (at 70°F)	31.3
Normal Condition	65.8
With 1% rods rupture	66.4
With 10% rods rupture	72.5
With 100% rods rupture	133.5

<sup>†</sup> Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

Table 4.4.15

SUMMARY OF HI-STORM SYSTEM COMPONENT TEMPERATURES  
FOR LONG-TERM STORAGE (°F)

<b>Location</b>	<b>MPC-24</b>	<b>MPC-68</b>	<b>MPC-32</b>	<b>MPC-24E</b>
MPC Basket Top:				
Basket periphery	485	501	496	488
MPC shell	344	348	351	346
Overpack Inner Shell	199	199	199	199
Overpack Outer Shell	124	124	124	124
MPC Basket Bottom:				
Basket periphery	281	280	290	284
MPC shell	256	258	261	258
Overpack Inner Shell	106	106	106	106
Overpack Outer Shell	107	107	107	107



Table 4.4.16

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Table 4.4.17

INTENTIONALLY DELETED

Table 4.4.18

INTENTIONALLY DELETED

Table 4.4.19

SUMMARY OF MPC CONFINEMENT BOUNDARY  
TEMPERATURE DISTRIBUTIONS

<b>Location</b>	<b>MPC-24 (°F)</b>	<b>MPC-68 (°F)</b>	<b>MPC-32 (°F)</b>	<b>MPC-24E (°F)</b>
MPC Lid Inside Surface at Centerline	463	502	487	462
MPC Lid Outside Surface at Centerline	427	454	447	425
MPC Lid Inside Surface at Periphery	371	381	383	372
MPC Lid Outside Surface at Periphery	360	375	372	358
MPC Baseplate Inside Surface at Centerline	207	209	214	209
MPC Baseplate Outside Surface at Centerline	200	203	208	202
MPC Baseplate Inside Surface at Periphery	243	246	249	245
MPC Baseplate Outside Surface at Periphery	194	196	199	195

Table 4.4.20

MPC-24 DESIGN-BASIS MAXIMUM HEAT LOAD<sup>†</sup>  
 VERSUS FUEL AGE AT LOADING (MODERATE BURNUP)

Fuel Age At Loading (years)	Permissible Heat Load (kW)
5	27.77
6	26.96
7	24.74
10	24.23
15	23.66

<sup>†</sup> The cask heat load limits ( $Q_{\tau}$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_{\tau}$ ), where “ $\tau$ ” is the age of the fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of  $\tau$ ) as specified in the Appendix B to COC 1014.

Table 4.4.21

MPC-68 DESIGN-BASIS MAXIMUM HEAT LOAD<sup>†</sup>  
 VERSUS FUEL AGE AT LOADING (MODERATE BURNUP)

Fuel Age At Loading (years)	Permissible Heat Load (kW)
5	28.19
6	26.81
7	24.71
10	24.18
15	23.60

<sup>†</sup> The cask heat load limits ( $Q_c$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_c$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.22  
MATRIX OF HI-STORM SYSTEM THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Long Term Normal	Ambient	SS	$N_T, Q_D, ST, SC, I_O$	4.4.2
2	Off-Normal Environment	Ambient	SS(B)	$O_T, Q_D, ST, SC, I_O$	11.1.2
3	Extreme Environment	Ambient	SS(B)	$E_T, Q_D, ST, SC, I_O$	11.2.15
4	Partial Ducts Blockage	Ambient	SS(B)	$N_T, Q_D, ST, SC, I_{1/24}$	11.1.4
5	Ducts Blockage Accident	Overpack	TA	$N_T, Q_D, ST, SC, I_C$	11.2.13
6	Fire Accident	Overpack	TA	$Q_D, F$	11.2.4
7	Tip Over Accident	Overpack	AH	$Q_D$	11.2.3
8	Debris Burial Accident	Overpack	AH	$Q_D$	11.2.14

Legend:

<p><math>N_T</math> - Maximum Annual Average (Normal) Temperature (80°F)  <math>O_T</math> - Off-Normal Temperature (100°F)  <math>E_T</math> - Extreme Hot Temperature (125°F)  <math>Q_D</math> - Design Basis Maximum Heat Load  SS - Steady State  SS(B) - Bounding Steady State  TA - Transient Analysis  AH - Adiabatic Heating</p>	<p><math>I_O</math> - All Inlet Ducts Open  <math>I_{1/2}</math> - Half of Inlet Ducts Open  <del><math>I_{1/4}</math> - Quarter of Inlet Ducts Open</del>    <math>I_C</math> - All Inlet Ducts Closed    ST - Insolation Heating (Top)  SC - Insolation Heating (Curved)  F - Fire Heating (1475°F)</p>
---	---

Table 4.4.23

## PLANT SPECIFIC BWR FUEL TYPES EFFECTIVE CONDUCTIVITY†

<b>Fuel</b>	<b>@200°C [Btu/ft-hr-°F]</b>	<b>@450°F [Btu/ft-hr-°F]</b>	<b>@700°F [Btu/ft-hr-°F]</b>
Oyster Creek (7x7)	0.161	0.269	0.422
Oyster Creek (8x8)	0.162	0.266	0.413
TVA Browns Ferry (8x8)	0.160	0.264	0.411
SPC-5 (9x9)	0.149	0.245	0.380
ANF 8x8	0.167	0.277	0.433
ANF-9X (9x9)	0.165	0.272	0.423

---

† The conductivities reported in this table are obtained by a simplified analytical method in Subsection 4.4.1.1.2.



Table 4.4.24

SUMMARY OF MPC-32 FREE VOLUME CALCULATIONS

Item	Volume (ft <sup>3</sup> )
Cavity Volume	367.9
Basket Metal Volume	27.4
Bounding Free Assemblies Volume	105.0
Basket Supports and Fuel Spacers Volume	9.0
<del>Optional Aluminum Conduction Elements</del>	<del>5.9</del>
Net Free Volume	<del>220.6 (6,247 liters)</del> 226.5 (6,414 liters)*

---

\* A conservative lowerbound value of 220.6 ft<sup>3</sup> (6,247 liters) is used for subsequent MPC internal pressure calculations.

Table 4.4.25

SUMMARY OF MPC-24E FREE VOLUME CALCULATIONS

Item	Volume (ft <sup>3</sup> )
Cavity Volume	367.9
Basket Metal Volume	51.2
Bounding Fuel Assemblies Volume	78.8
Basket Supports and Fuel Spacers Volume	6.1
<del>Optional Aluminum Conduction Elements</del>	<del>5.9</del>
Net Free Volume	<del>225.9 (6,398 liters)</del> 231.8 (6,564 liters)*

---

\* A conservative lowerbound value of 225.9 ft<sup>3</sup> (6,398 liters) is used for subsequent MPC internal pressure calculations.

Table 4.4.26

HI-STORM<sup>†</sup> SYSTEM LONG-TERM NORMAL STORAGE MAXIMUM TEMPERATURES  
(MPC-32 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	691	752787 <sup>††</sup>
MPC Basket	660	725 <sup>†††</sup>
Basket Periphery	496	725 <sup>†††</sup>
MPC Outer Shell	351	450

<sup>†</sup> Bounding overpack temperatures are provided in Table 4.4.36.

<sup>††</sup> ~~This temperature limit is in accordance with ISG-11 [4.1.4]. The temperature limit is in accordance with DCCG (gross rupture) criteria. Permissible peak cladding temperature is 691°F PNL Criteria).~~

<sup>†††</sup> The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism in the analysis of the HI-STORM Systems.

Table 4.4.27

HI-STORM<sup>†</sup> SYSTEM LONG-TERM NORMAL STORAGE MAXIMUM TEMPERATURES  
(MPC-24E BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	691	752787 <sup>††</sup>
MPC Basket	650	725 <sup>†††</sup>
Basket Periphery	492	725 <sup>†††</sup>
MPC Outer Shell	347	450

<sup>†</sup> Bounding overpack temperatures are provided in Table 4.4.36.

<sup>††</sup> ~~This temperature limit is in accordance with ISG-11 [4.1.4]. The temperature limit is in accordance with DCCG (gross rupture) criteria. Permissible peak cladding temperature is 691°F (PNL Criteria).~~

<sup>†††</sup> The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism to the analysis of the HI-STORM System.

Table 4.4.28

MPC-32 DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup> VERSUS FUEL AGE AT LOADING  
(MODERATE BURNUP)

Fuel Age at Loading (years)	Permissible Heat Load (kW)
5	28.74
6	27.95
7	25.79
10	25.26
15	24.68

<sup>†</sup> The cask heat load limits ( $Q_{\tau}$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_{\tau}$ ) where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to CoC 1014.

Table 4.4.29

MPC-24E DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup> VERSUS FUEL AGE AT LOADING  
(MODERATE BURNUP)

Fuel Age at Loading (years)	Permissible Heat Load (kW)
5	28.17
6	27.33
7	25.05
10	24.53
15	23.95

<sup>†</sup> The cask heat load limits ( $Q_c$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_c$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.30

REGIONALIZED LOADING OUTER REGION HEAT LOAD LIMITS

<b>MPC Type</b>	<b>Inner Region Assemblies</b>	<b>Outer Region Assemblies</b>	<b>Outer Region Heat Load (kW)</b>
MPC-24	4	20	18
MPC-24E	4	20	18
MPC-32	12	20	12
MPC-68	32	36	9.9

Table 4.4.31

REGIONALIZED LOADING INNER REGION HEAT LOAD LIMITS (kW)

<b>Fuel Age (years)</b>	<b>MPC-24</b>	<b>MPC-24E</b>	<b>MPC-32</b>	<b>MPC-68</b>
5	5.88 <sup>†</sup>	6.16 <sup>†</sup>	13.58	16.02
6	5.88 <sup>‡</sup>	6.16 <sup>‡</sup>	12.87	14.99
7	5.34	5.58	11.92	13.40
10	4.94	5.16	11.40	12.99
15	4.66	4.86	11.02	12.54

<sup>†</sup> Inner region heat load governed by interface cladding temperature limit.



Table 4.4.32

~~[INTENTIONALLY DELETED] MPC-24 DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup>  
VERSUS FUEL AGE  
AT LOADING (HIGH BURNUP)~~

Fuel Age at Loading (yrs)	Permissible Heat Load (kW)
5	27.12
6	26.09
7	24.74
10	24.02
15	23.50

<sup>†</sup> The cask heat load limits ( $Q_{\tau}$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_{\tau}$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.33

~~[INTENTIONALLY DELETED] MPC-24E DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup> VERSUS FUEL AGE AT LOADING (HIGH BURNUP)~~

Fuel Age at Loading (yrs)	Permissible Heat Load (kW)
5	27.50
6	26.44
7	25.05
10	24.31
15	23.79

<sup>†</sup> The cask heat load limits ( $Q_c$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_c$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.34

~~[INTENTIONALLY DELETED] MPC-32 DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup> VERSUS FUEL AGE AT LOADING (HIGH BURNUP)~~

Fuel Age at Loading (yrs)	Permissible Heat Load (kW)
5	28.10
6	27.10
7	25.79
10	25.05
15	24.53

<sup>†</sup> The cask heat load limits ( $Q_c$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_c$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.35

~~[INTENTIONALLY DELETED] MPC-68 DESIGN BASIS MAXIMUM HEAT LOAD<sup>†</sup> VERSUS FUEL AGE AT LOADING (HIGH BURNUP)~~

Fuel Age at Loading (yrs)	Permissible Heat Load (kW)
5	28.19
6	26.81
7	24.71
10	24.18
15	23.60

<sup>†</sup> The cask heat load limits ( $Q_c$ ) presented in this table pertain to loading the MPC with uniformly aged fuel assemblies emitting heat at the design basis maximum rate ( $q_c$ ), where “ $\tau$ ” is the age of fuel at the start of dry storage. For a cask loaded with a mix of fuel ages, the cask heat load limit shall be the sum of the individual assembly decay heat limits (as a function of fuel age) as specified in the Appendix B to COC 1014.

Table 4.4.36

BOUNDING LONG-TERM NORMAL STORAGE  
HI-STORM OVERPACK TEMPERATURES

Component <sup>†</sup>	Local Section Temperature <sup>††</sup> (°F)	Long-Term Temperature Limit (°F)
Inner shell	199	350
Outer shell	145	350
Lid bottom plate	339	350
Lid top plate	196	350
MPC pedestal plate	208	350
Baseplate	111	350
Radial shield	172	200
Air outlet <sup>†††</sup>	206	

<sup>†</sup> See Figure 1.2.8 for a description of HI-STORM components.

<sup>††</sup> Section temperature is defined as the through-thickness average temperature.

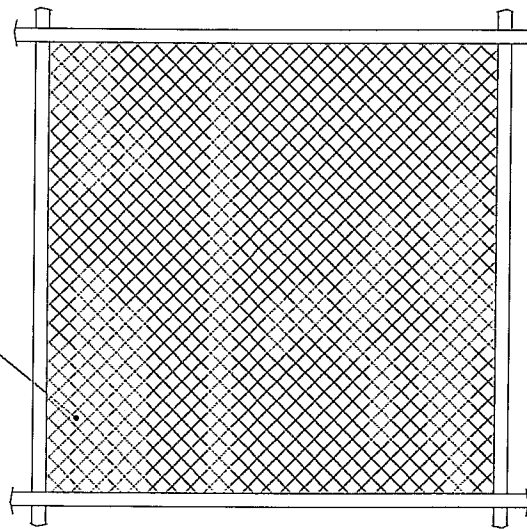
<sup>†††</sup> Reported herein for the option of temperature measurement surveillance of outlet ducts air temperature as set forth in the Technical Specifications.

Table 4.4.37

*SUMMARY OF HI-STORM DIFFERENTIAL THERMAL EXPANSIONS*

<i>Gap Description</i>	<i>Cold Gap U (in)</i>	<i>Differential Expansion V (in)</i>	<i>Is Free Expansion Criterion Satisfied (i.e., U &gt; V)</i>
<i>Fuel Basket-to-MPC Radial Gap</i>	<i>0.1875</i>	<i>0.096</i>	<i>Yes</i>
<i>Fuel Basket-to-MPC Axial Gap</i>	<i>1.25</i>	<i>0.499</i>	<i>Yes</i>
<i>MPC-to-Overpack Radial Gap</i>	<i>0.5</i>	<i>0.139</i>	<i>Yes</i>
<i>MPC-to-Overpack Axial Gap</i>	<i>1.0</i>	<i>0.771</i>	<i>Yes</i>

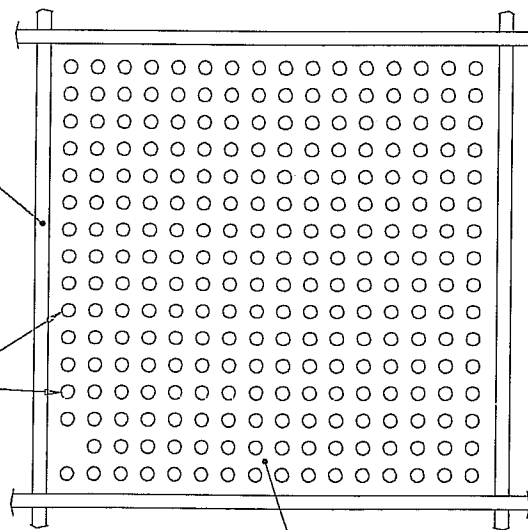
HOMOGENEOUS CROSS-SECTION  
WITH UNIFORM HEAT GENERATION



(b) SOLID REGION OF  
EFFECTIVE CONDUCTIVITY

HEAT GENERATING  
FUEL RODS ARRAY

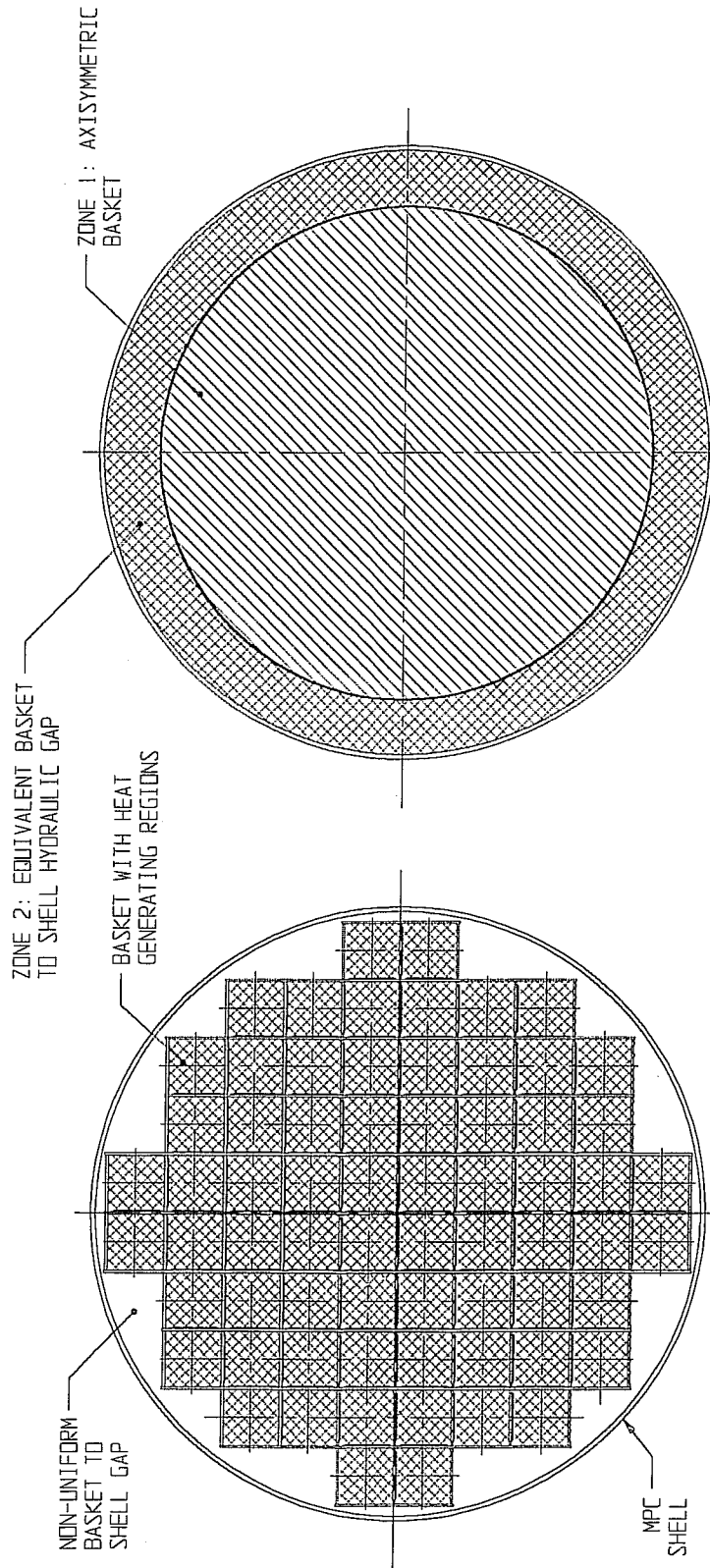
BASKET CELL WALL



HELIUM FILLING  
EMPTY SPACES

(a) TYPICAL FUEL CELL

FIGURE 4.4.1; HOMOGENIZATION OF THE STORAGE CELL CROSS-SECTION

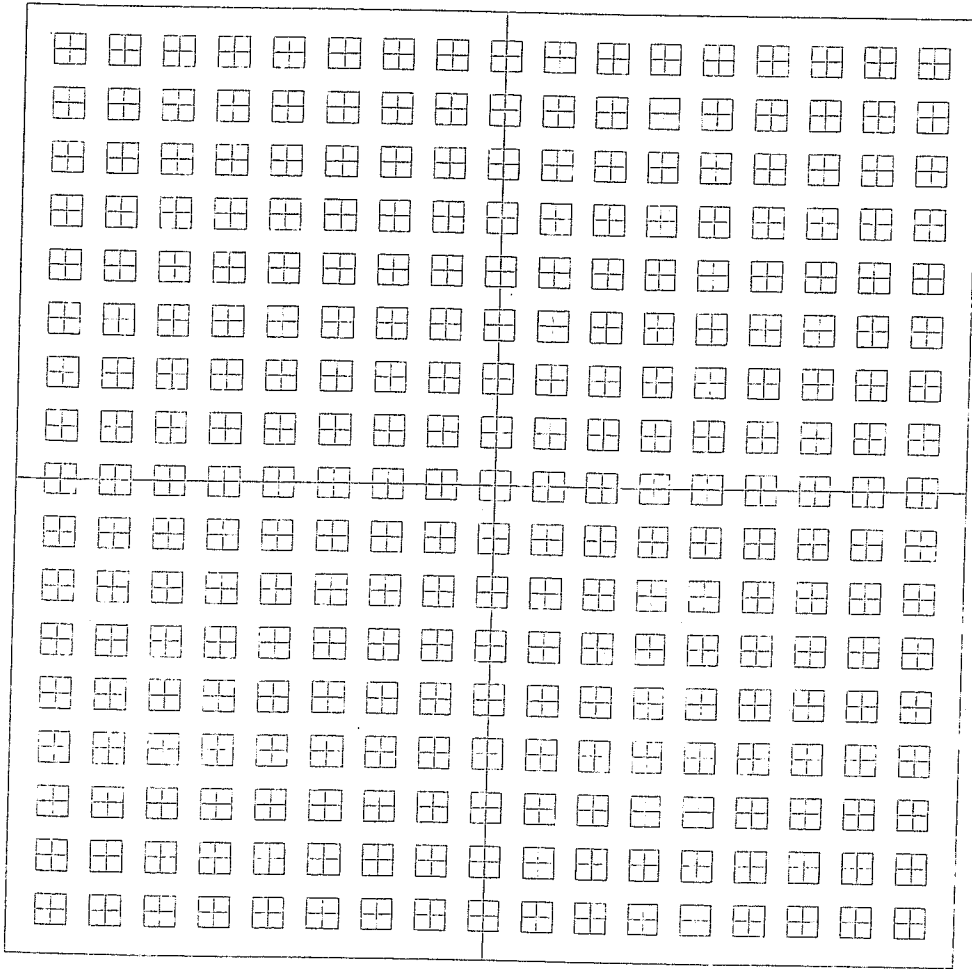


( a ) TYPICAL MPC CROSS-SECTION

( b ) EQUIVALENT TWO ZONE MODEL

FIGURE 4.4.2: MPC CROSS-SECTION REPLACED WITH AN EQUIVALENT TWO ZONE AXISYMMETRIC BODY

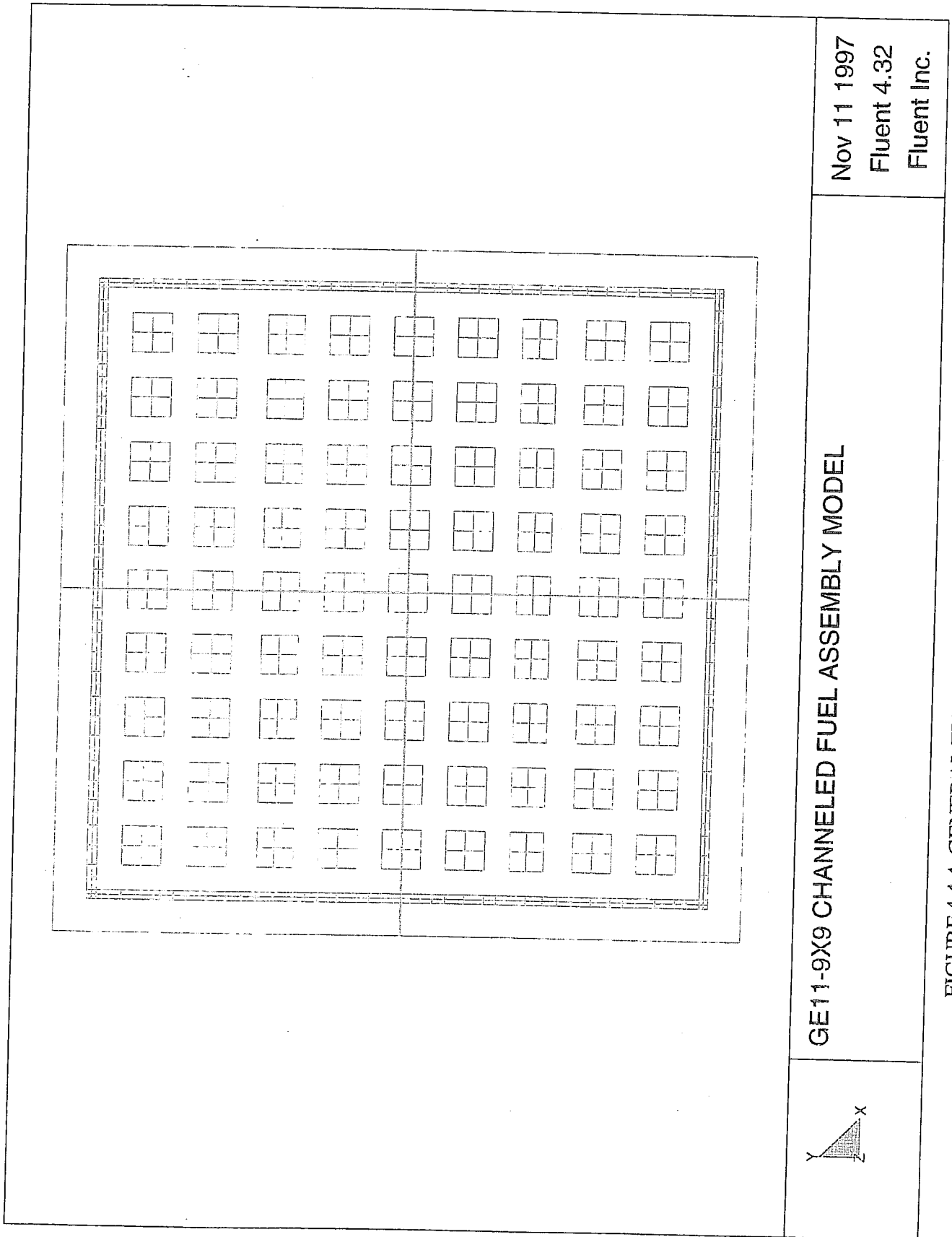




W17X17 OFA FUEL ASSEMBLY MODEL

Nov 11 1997  
 Fluent 4.32  
 Fluent Inc.

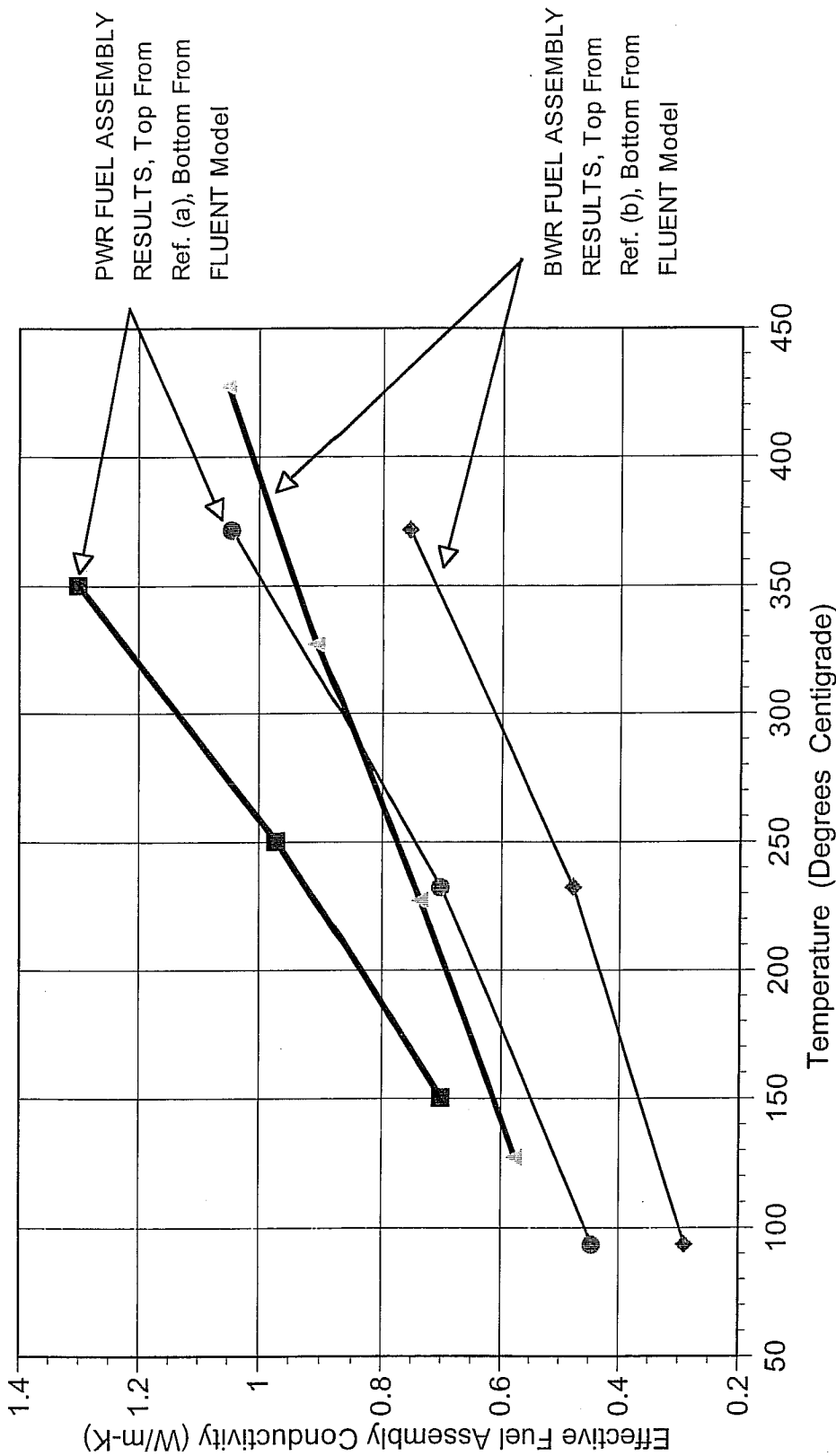
FIGURE 4.4.3: WESTINGHOUSE 17x17 OFA PWR FUEL ASSEMBLY MODEL



GE11-9X9 CHANNELED FUEL ASSEMBLY MODEL

Nov 11 1997  
 Fluent 4.32  
 Fluent Inc.

FIGURE 4.4.4: GENERAL ELECTRIC 9x9 BWR FUEL ASSEMBLY MODEL



(a) "Determination of SNF Peak Temperatures in the Waste Package", Bahney & Doering, HLWWM Sixth Annual Conf., Pages 671-673, (April 30 - May 5, 1995).  
(b) "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements", Sandia Report SAND90-2406, page II-132, (1992).

FIGURE 4.4.5; COMPARISON OF FLUENT CALCULATED FUEL ASSEMBLY CONDUCTIVITY RESULTS WITH PUBLISHED TECHNICAL DATA

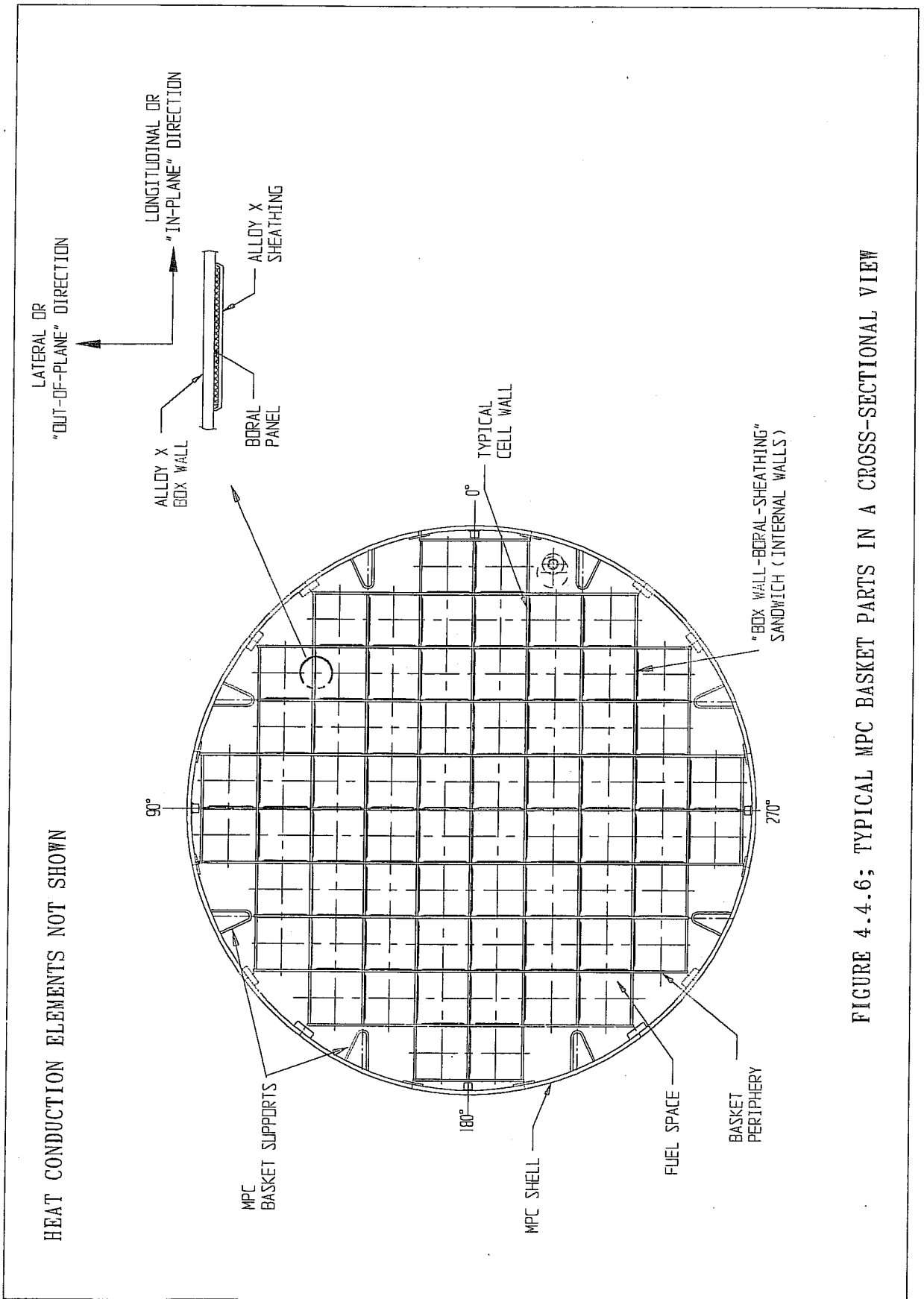


FIGURE 4.4.6; TYPICAL MPC BASKET PARTS IN A CROSS-SECTIONAL VIEW

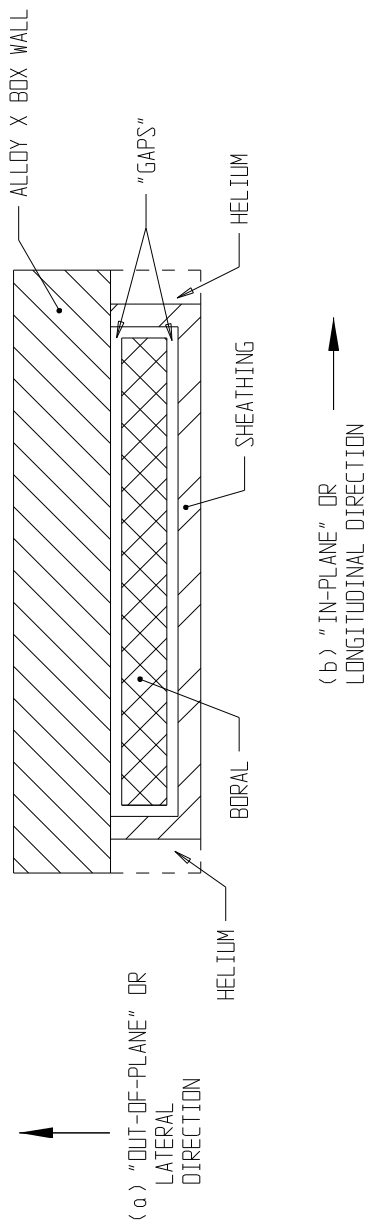


FIGURE 4.4.7; "BOX WALL-BORAL-SHEATHING" SANDWICH

FIGURE 4.4.8

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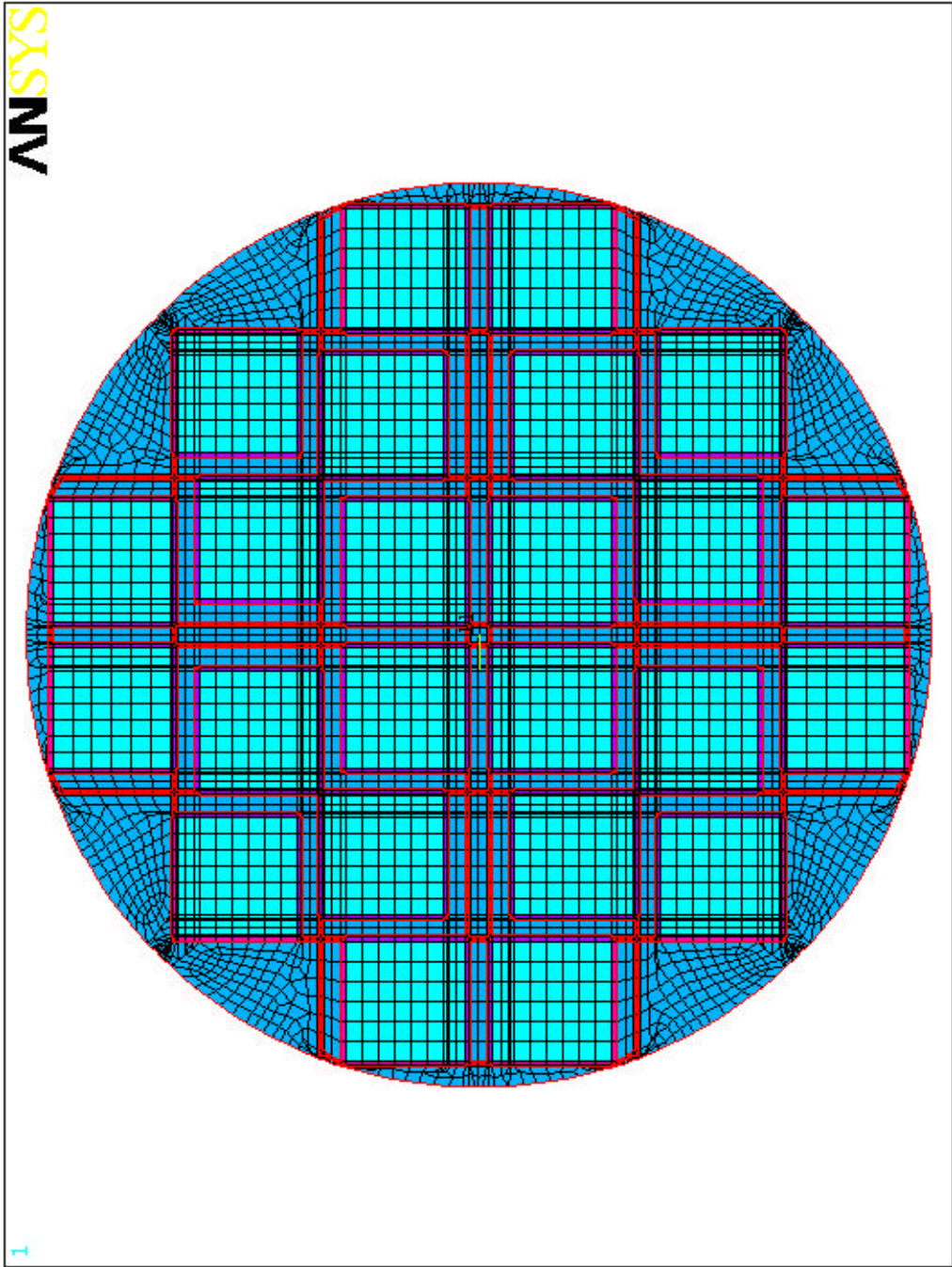


FIGURE 4.4.9; MPC-24 BASKET CROSS-SECTION ANSYS FINITE-ELEMENT MODEL

ANSYS 5.3  
NOV 13 1997  
11:28:39  
PLOT NO. 1  
ELEMENTS  
MAT NUM  
ZV =1  
\*DIST=37.606  
Z-BUFFER

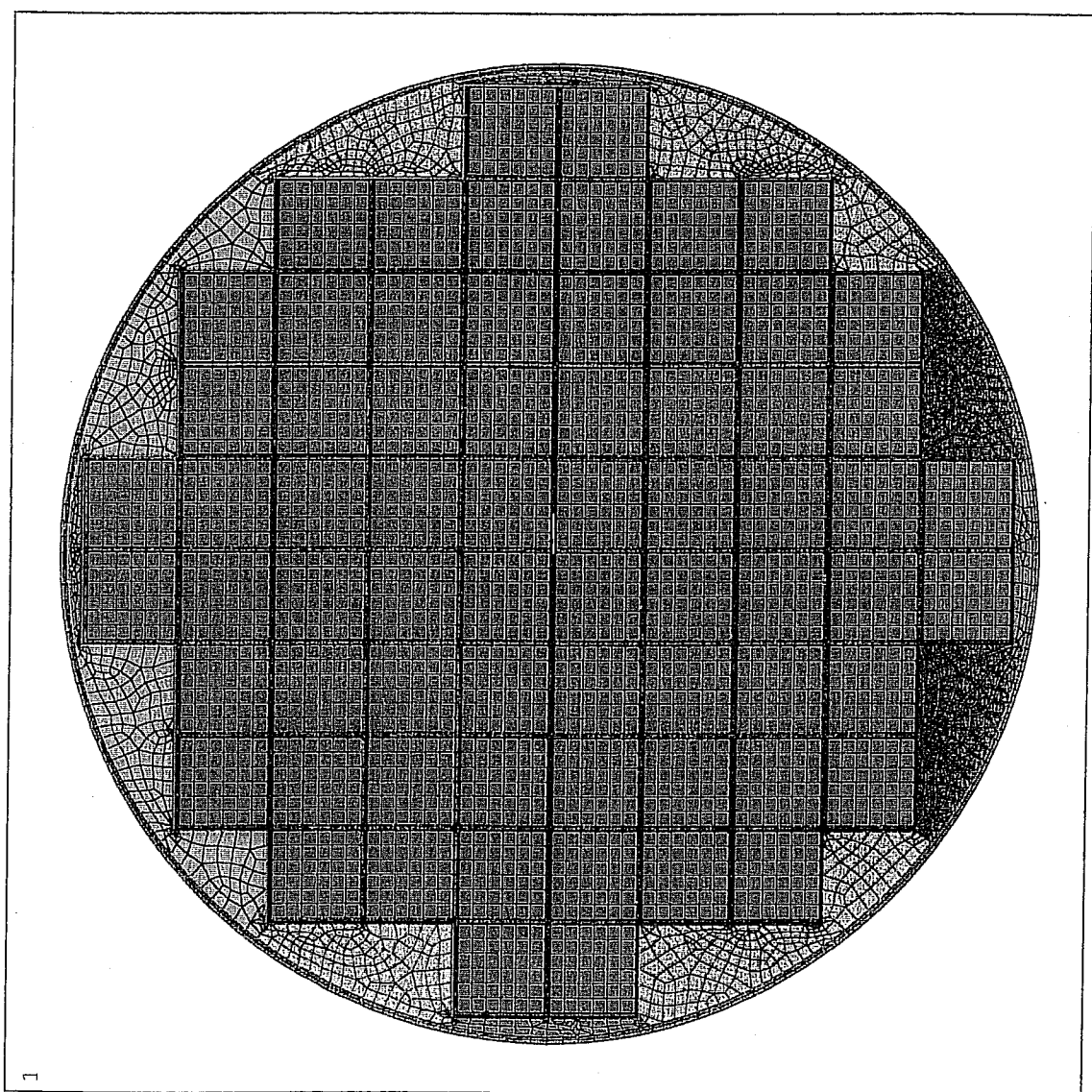
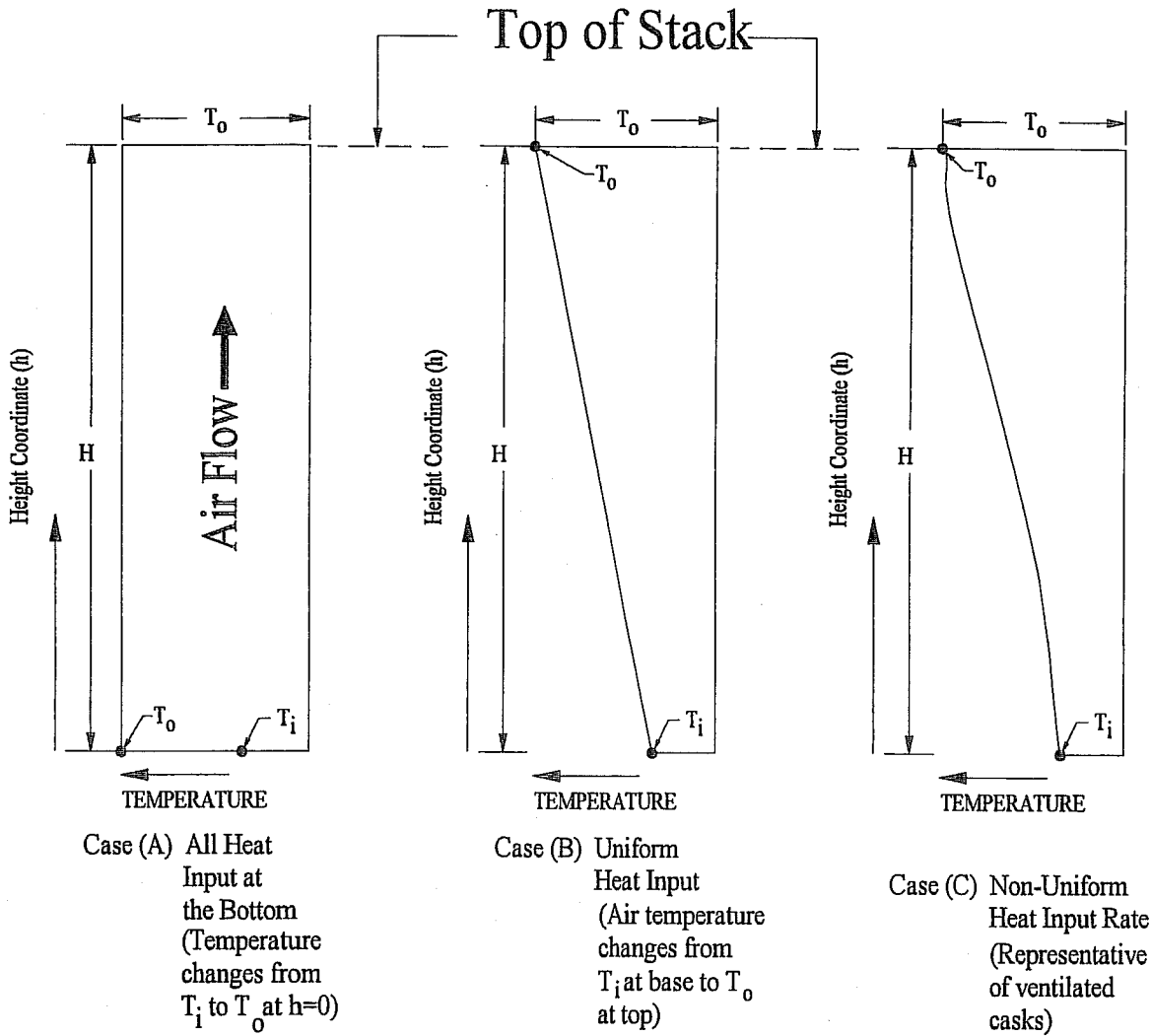


FIGURE 4.4.10; MPC-68 BASKET CROSS-SECTION ANSYS FINITE ELEMENT MODEL



FIGURE 4.4.11

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**Figure 4.4.12; Stack Air Temperature as a Function of Height**

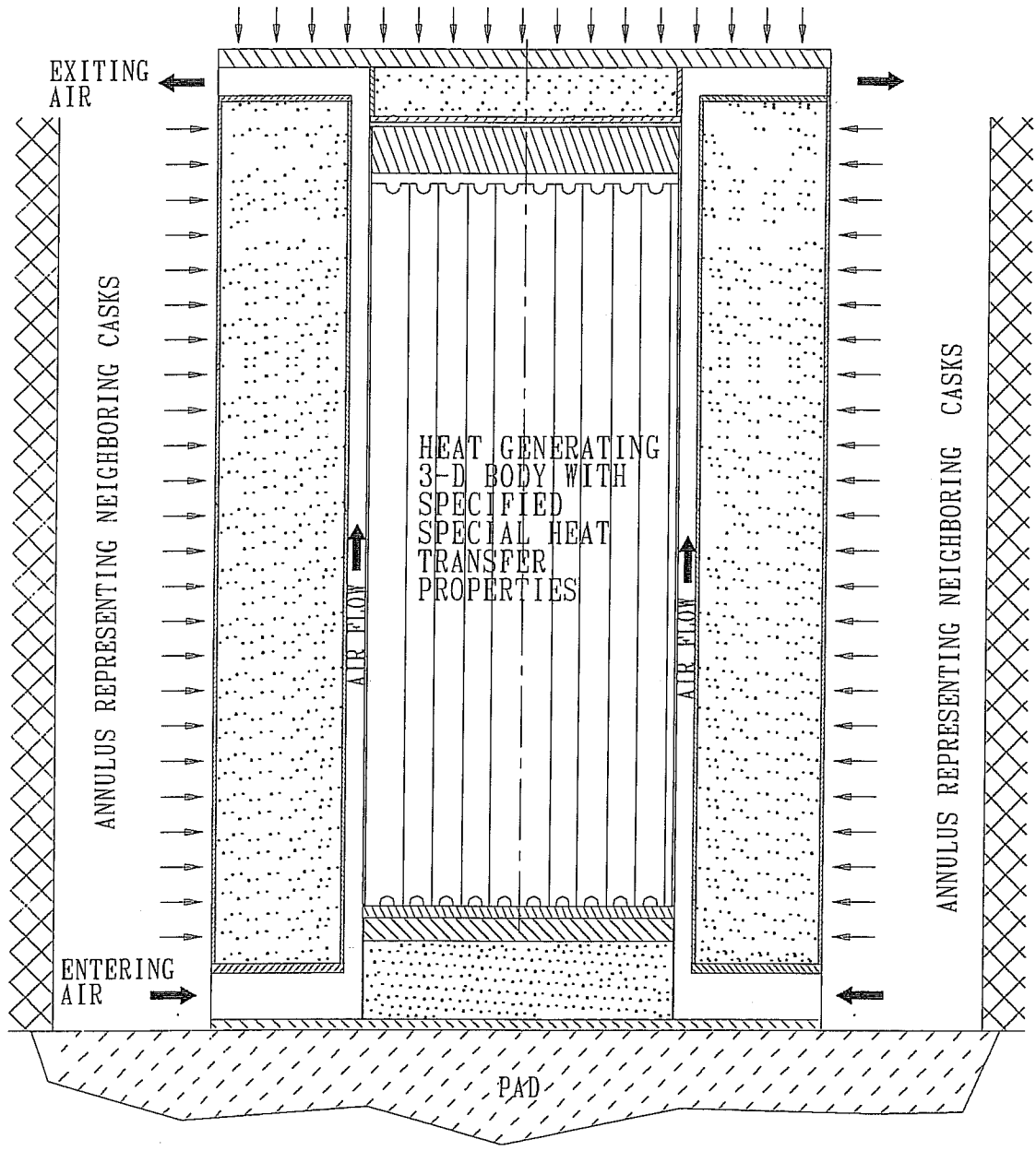
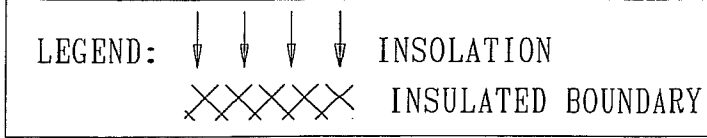


FIGURE 4.4.13; SCHEMATIC DEPICTION OF THE HI-STORM THERMAL ANALYSIS

**FIGURE 4.4.14**  
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FIGURE 4.4.15

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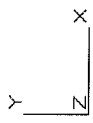
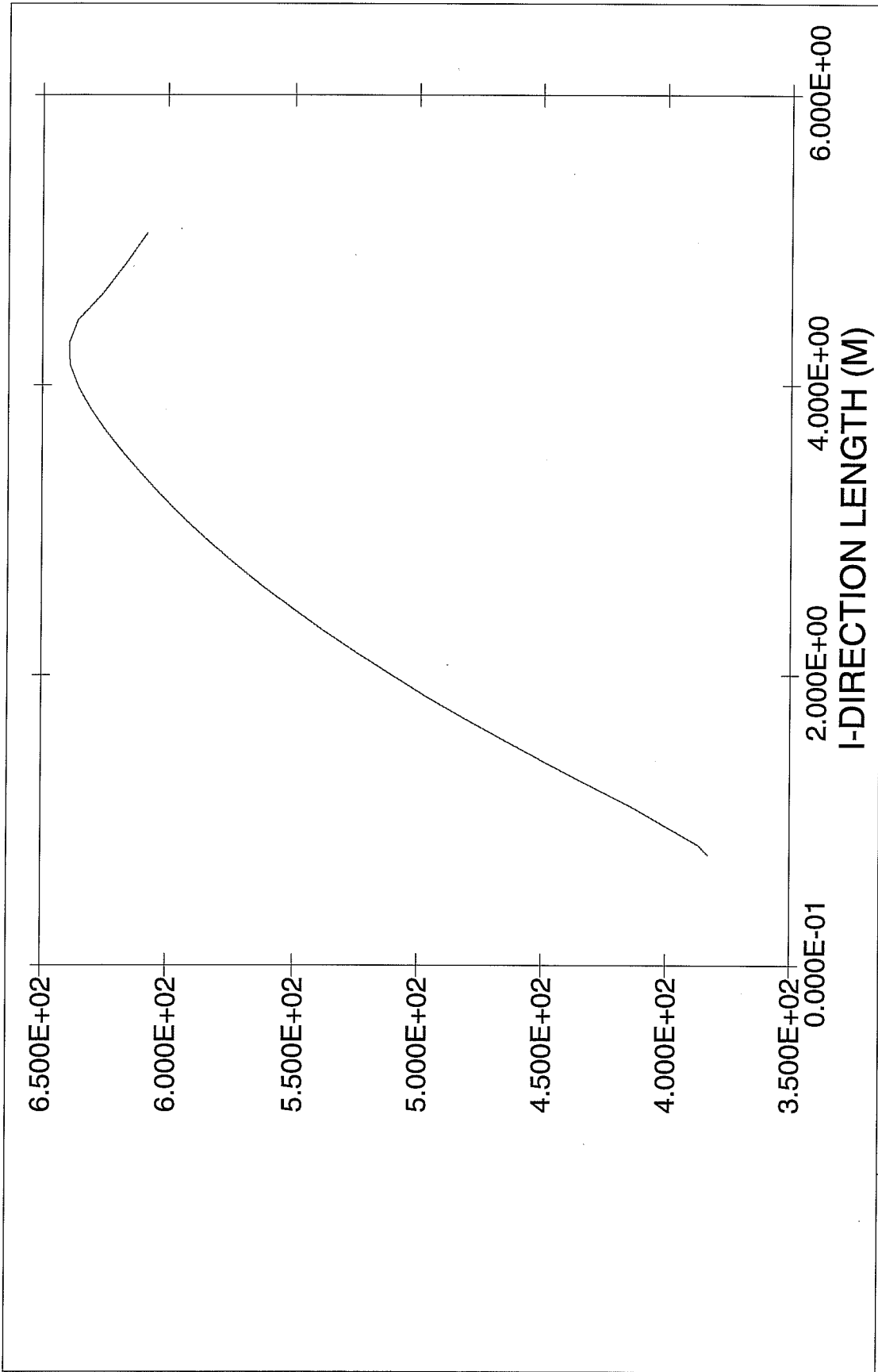


FIGURE 4.4.16: MPC-24 Peak Rod Axial Temperature Profile Aug 22 2000  
 Fluent 4.48  
 Fluent Inc.

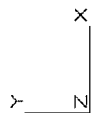
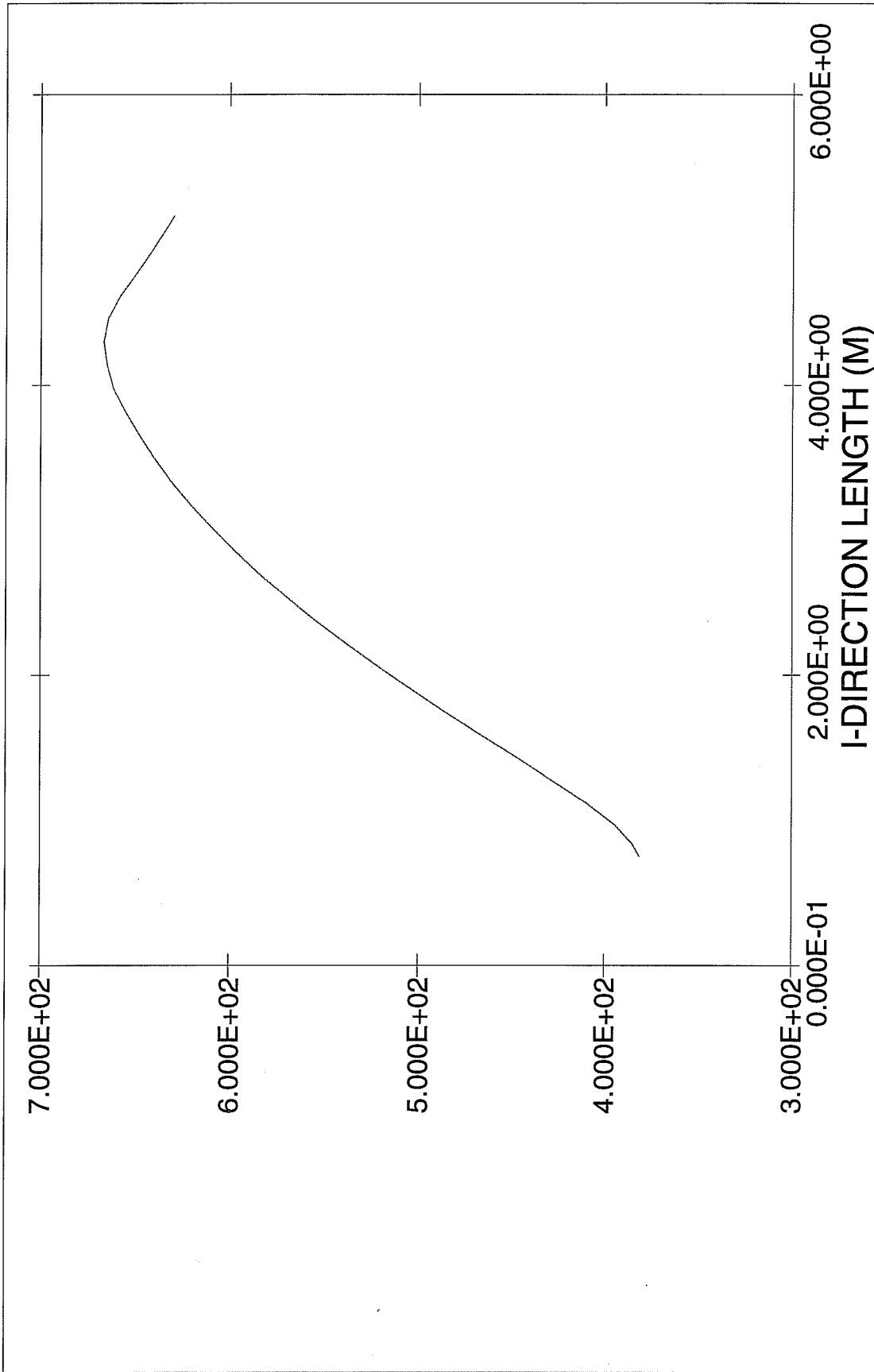


FIGURE 4.4.17: MPC-68 Peak Rod Axial Temperature Profile Aug 22 2000  
 Fluent 4.48  
 Fluent Inc.

FIGURE 4.4.18

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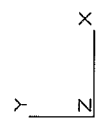
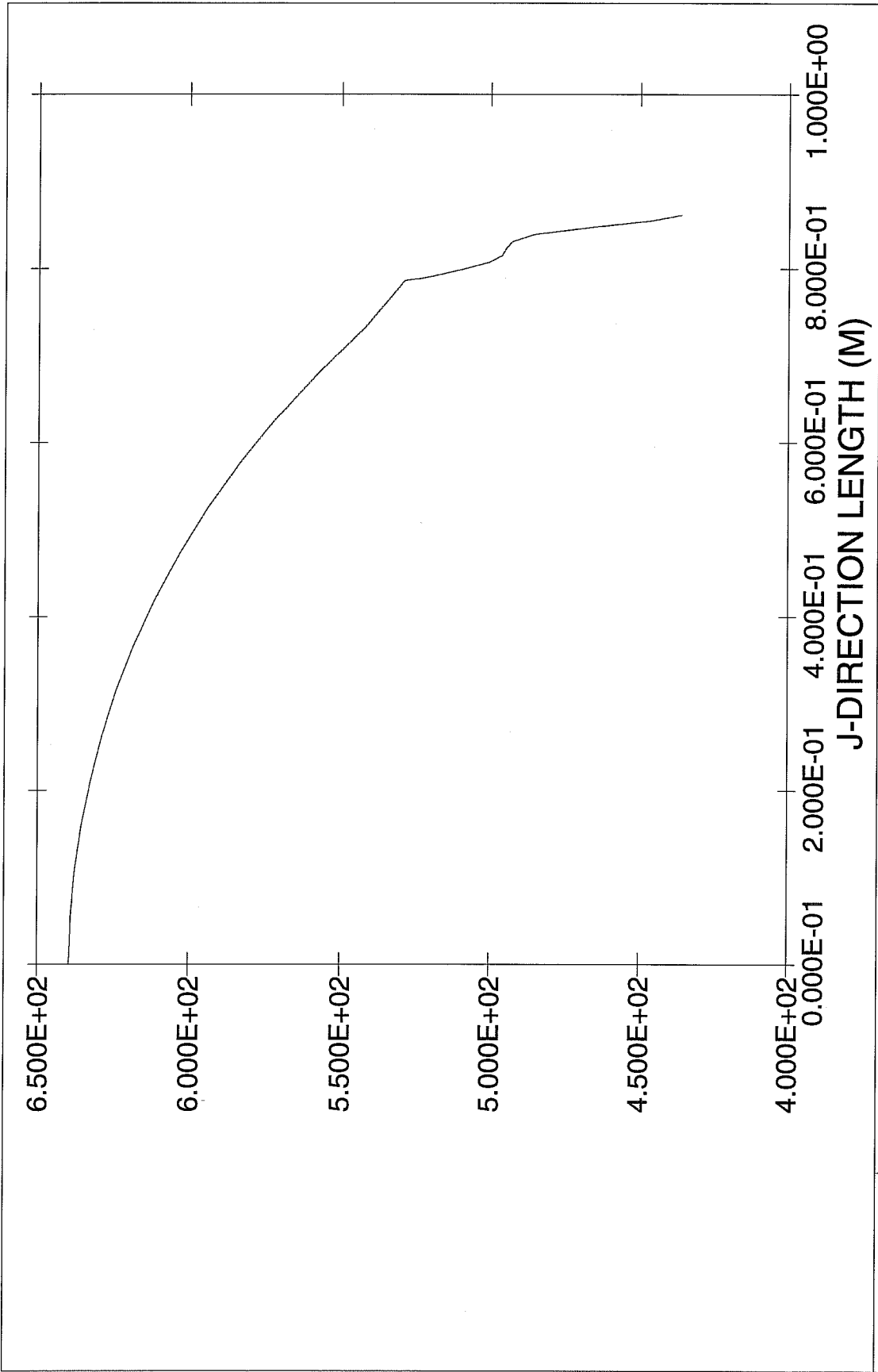


FIGURE 4.4.19: MPC-24 Radial Temperature Profile  
(Hottest Basket Cross-Section)

Aug 23 2000  
Fluent 4.48  
Fluent Inc.

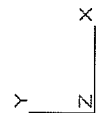
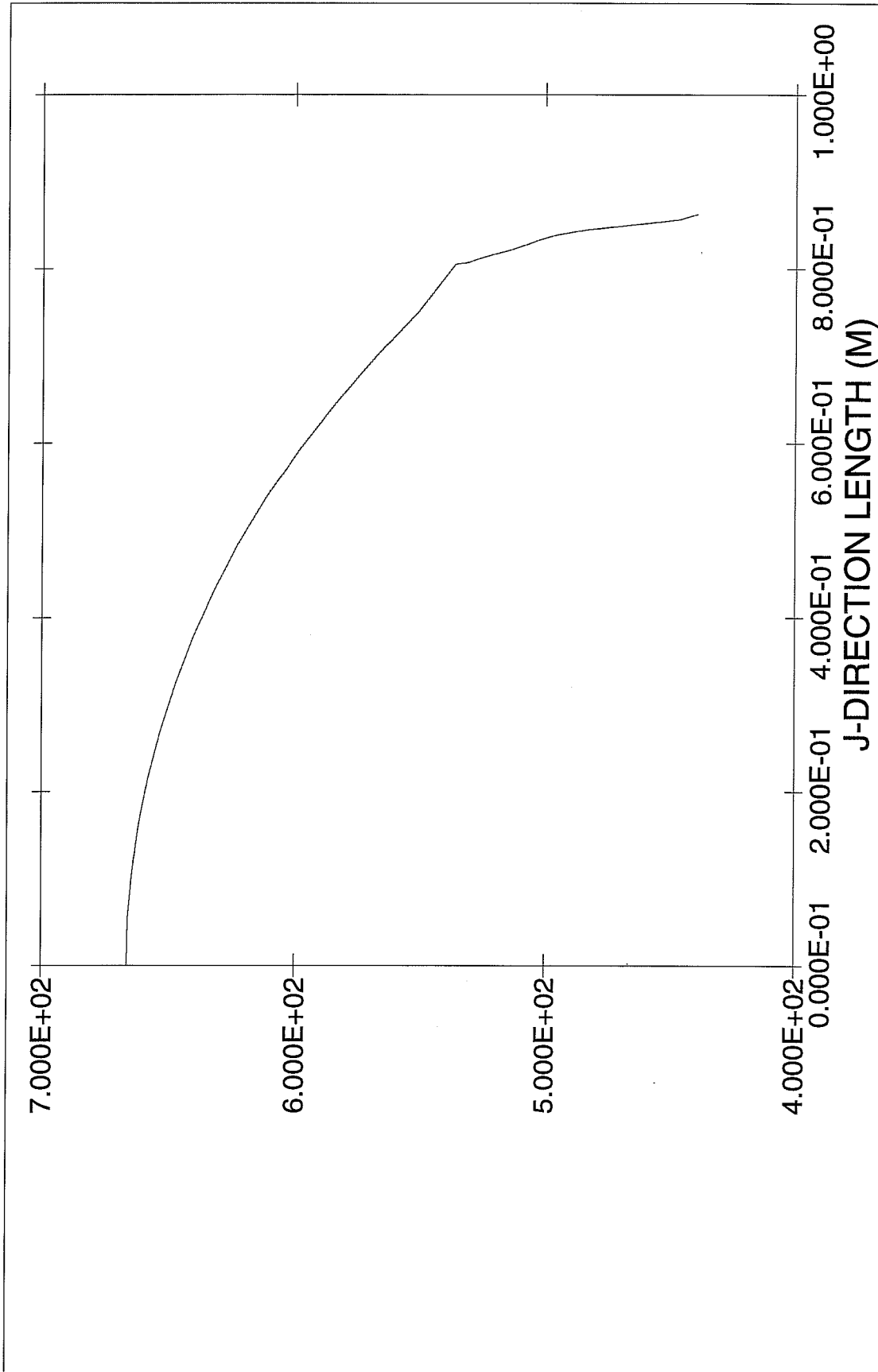


FIGURE 4.4.20: MPC-68 Radial Temperature Profile  
(Hottest Basket Cross-section)

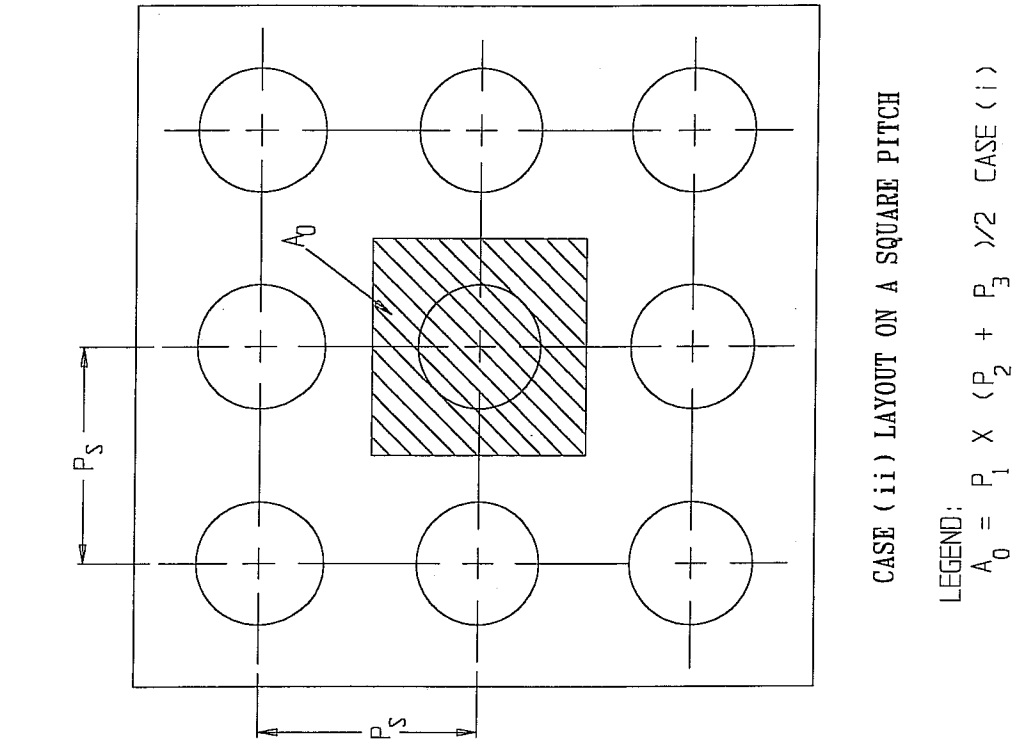
Aug 23 2000  
Fluent 4.48  
Fluent Inc.

FIGURE 4.4.21

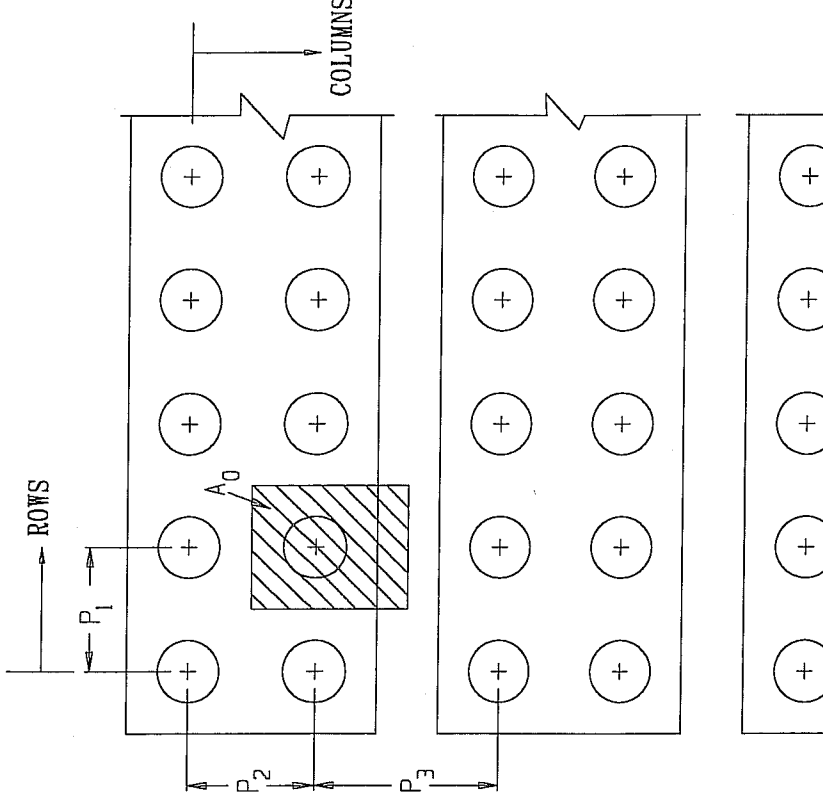
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**FIGURE 4.4.22**  
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**FIGURE 4.4.23**  
**INTENTIONALLY DELETED**



CASE (i) LAYOUT ON A RECTANGULAR PITCH



CASE (ii) LAYOUT ON A SQUARE PITCH

LEGEND:

$$A_0 = P_1 \times (P_2 + P_3) / 2 \quad \text{CASE (i)}$$

$$A_0 = P_s \times P_s \quad \text{CASE (ii)}$$

FIGURE 4.4.24; ILLUSTRATION OF MINIMUM AVAILABLE PLANAR AREA PER HI-STORM MODULE AT AN ISFSI.

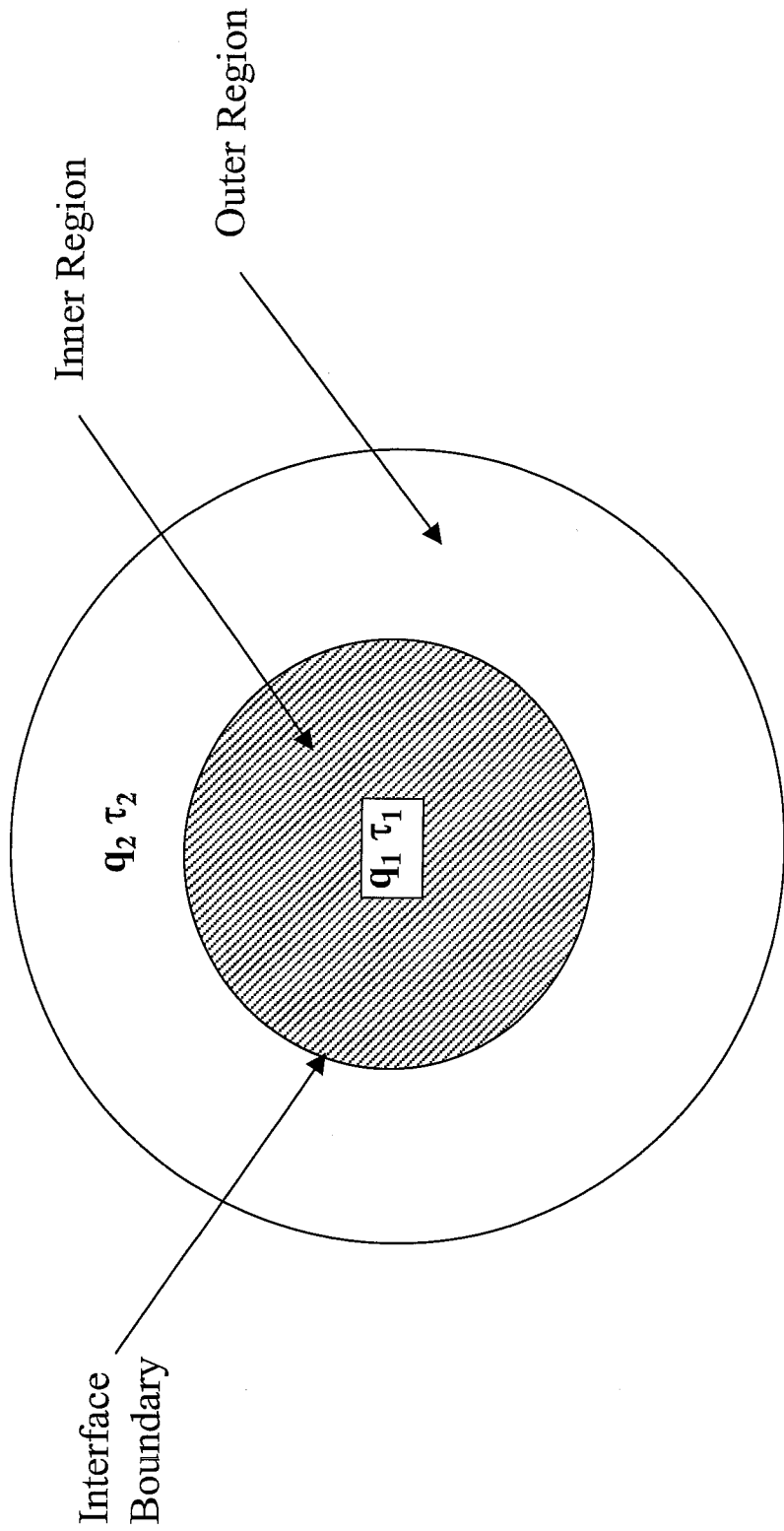


FIGURE 4.4.25: FUEL BASKET REGIONALIZED LOADING SCENARIO

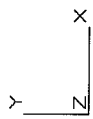
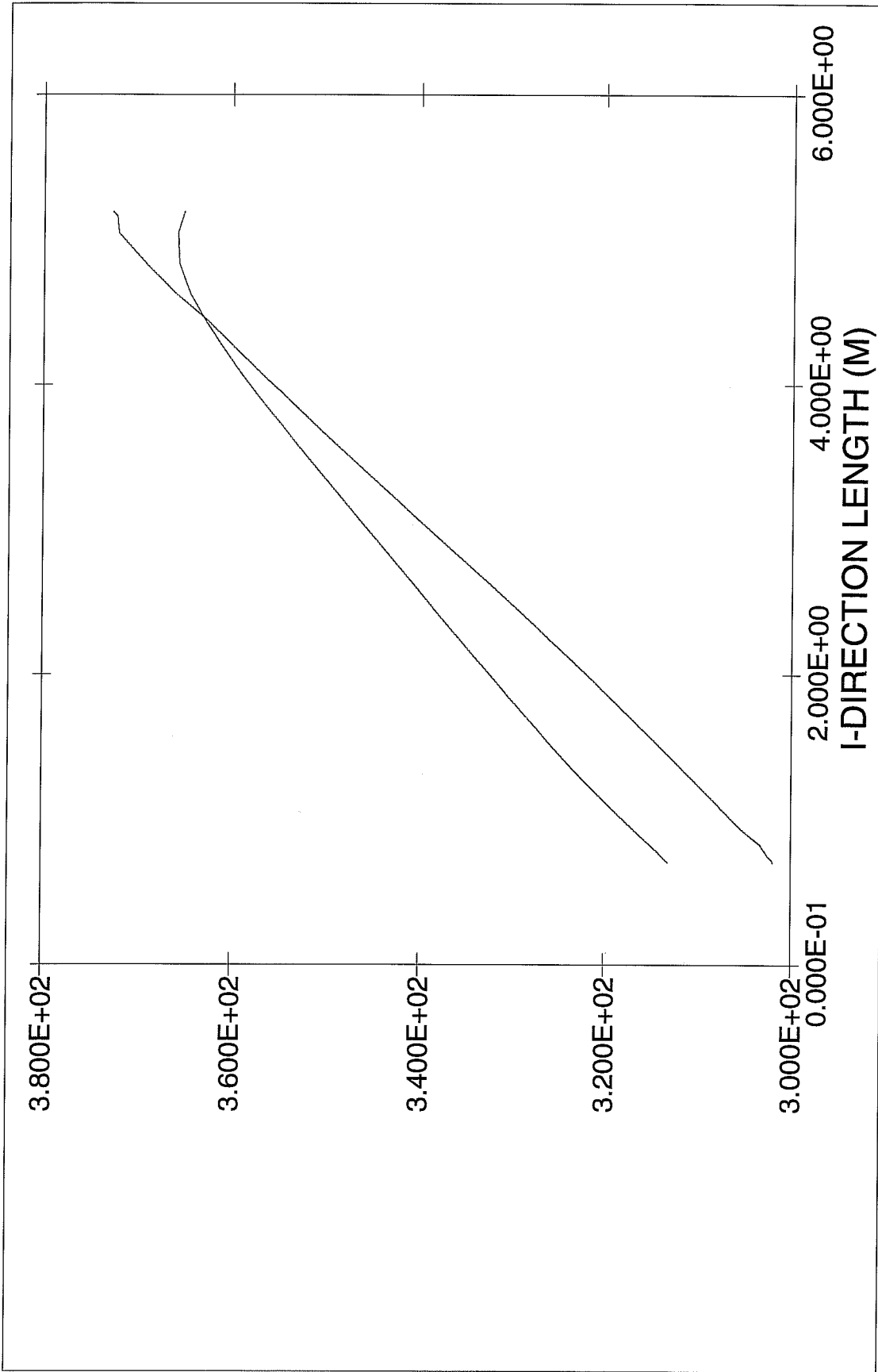


FIGURE 4.4.26: Bounding Overpack Annulus Axial Profiles

Aug 23 2000  
 Fluent 4.48  
 Fluent Inc.



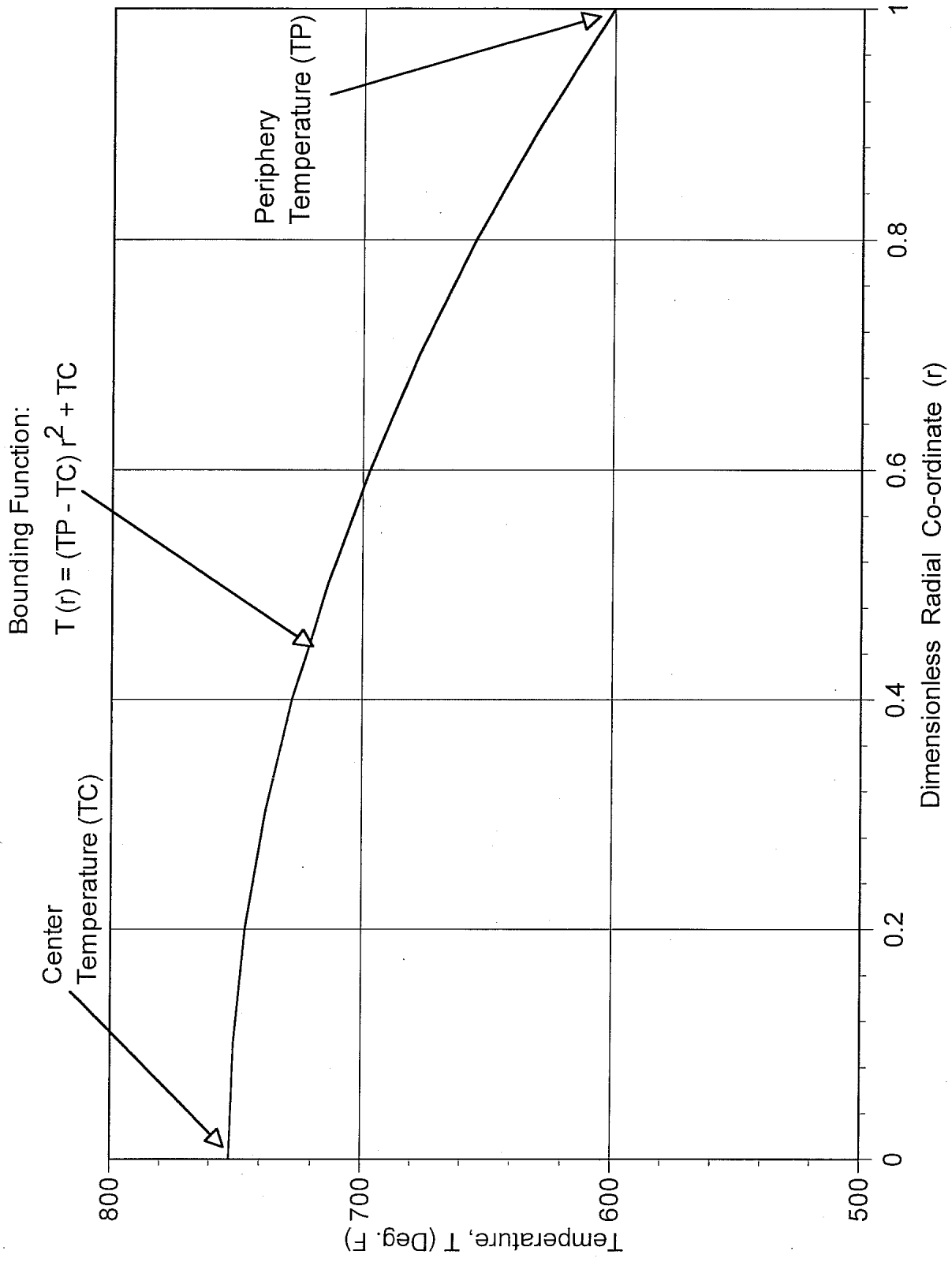


FIGURE 4.4.27: BOUNDING BASKET TEMPERATURE PROFILE FOR DIFFERENTIAL EXPANSION

#### 4.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS FOR NORMAL HANDLING AND ONSITE TRANSPORT

Prior to placement in a HI-STORM overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, ~~vacuum~~-dried, backfilled with helium and transported to the HI-STORM module. In the unlikely event that the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM overpacks or between a HI-STAR transport overpack and a HI-STORM storage overpack must be carried out in an assuredly safe manner. All of the above operations are short duration events that would likely occur no more than once or twice for an individual MPC.

The device central to all of the above operations is the HI-TRAC transfer cask that, as stated in Chapter 1, is available in two anatomically identical weight ratings (100- and 125-ton). The HI-TRAC transfer cask is a short-term host for the MPC; therefore it is necessary to establish that, during all thermally challenging operation events involving either the 100-ton or 125-ton HI-TRAC, the permissible temperature limits presented in Section 4.3 are not exceeded. The following discrete thermal scenarios, all of short duration, involving the HI-TRAC transfer cask have been identified as warranting thermal analysis.

- i. Normal Onsite Transport
- ii. MPC Cavity ~~Vacuum~~-Drying
- iii. Post-Loading Wet Transfer Operations
- iv. MPC Cooldown and Reflood for Unloading Operations

*Onsite transport of the MPC generally occurs with the HI-TRAC in the vertical orientation, which preserves the thermosiphon action within the MPC. However, there may be a scenario where onsite transport of an MPC must occur with the HI-TRAC in the horizontal configuration. Both orientations are evaluated in this section.*

*The fuel handling operations listed above place a certain level of constraint on the dissipation of heat from the MPC relative to the normal storage condition. Consequently, for some scenarios, it is necessary to provide additional cooling. For such situations, a new ancillary henceforth referred to as the Supplemental Cooling System (SCS) is required to provide additional cooling during short term operations. The specific design of an SCS must accord with site-specific needs and resources, including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the SCS will be satisfied by any plant-specific design are set forth in Appendix 2.C.*

The above listed conditions are described and evaluated in the following subsections. Subsection 4.5.1 describes the individual analytical models used to evaluate these conditions. Due to the simplicity of the conservative evaluation of wet transfer operations, Subsection 4.5.1.1.5 includes both the analysis model and analysis results discussions. The maximum temperature analyses for onsite transport and vacuum drying are discussed in Subsection 4.5.2. Subsections

4.5.3, 4.5.4 and 4.5.5, respectively, discuss minimum temperature, MPC maximum internal pressure and thermal data for stress analyses during onsite transport.

#### 4.5.1 Thermal Model

The HI-TRAC transfer cask is used to load and unload the HI-STORM concrete storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI pad. Section views of the HI-TRAC have been presented in Chapter 1. Within a loaded HI-TRAC, heat generated in the MPC is transported from the contained fuel assemblies to the MPC shell in the manner described in Section 4.4. From the outer surface of the MPC to the ambient air, heat is transported by a combination of conduction, thermal radiation and natural convection. ~~It has been demonstrated in Section 4.3 that from a thermal standpoint, storage of stainless steel clad fuel assemblies is bounded by storage of zircaloy clad fuel assemblies. Thus, only zircaloy clad fuel assemblies shall be considered in the HI-TRAC thermal performance evaluations.~~ Analytical modeling details of all the various thermal transport mechanisms are provided in the following subsection.

Two HI-TRAC transfer cask designs, namely, the 125-ton and the 100-ton versions, are developed for onsite handling and transport, as discussed in Chapter 1. The two designs are principally different in terms of lead thickness and the thickness of radial connectors in the water jacket region. The analytical model developed for HI-TRAC thermal characterization conservatively accounts for these differences by applying the higher shell thickness and thinner radial connectors' thickness to the model. In this manner, the HI-TRAC overpack resistance to heat transfer is overestimated, resulting in higher predicted MPC internals and fuel cladding temperature levels.

##### 4.5.1.1 Analytical Model

From the outer surface of the MPC to the ambient atmosphere, heat is transported within HI-TRAC through multiple concentric layers of air, steel and shielding materials. Heat must be transported across a total of six concentric layers, representing the air gap, the HI-TRAC inner shell, the lead shielding, the HI-TRAC outer shell, the water jacket and the enclosure shell. From the surface of the enclosure shell heat is rejected to the atmosphere by natural convection and radiation.

A small diametral air gap exists between the outer surface of the MPC and the inner surface of the HI-TRAC overpack. Heat is transported across this gap by the parallel mechanisms of conduction and thermal radiation. Assuming that the MPC is centered and does not contact the transfer overpack walls conservatively minimizes heat transport across this gap. Additionally, thermal expansion that would minimize the gap is conservatively neglected. Heat is transported through the cylindrical wall of the HI-TRAC transfer overpack by conduction through successive layers of steel, lead and steel. A water jacket, which provides neutron shielding for the HI-TRAC overpack, surrounds the cylindrical steel wall. The water jacket is composed of carbon steel channels with welded, connecting enclosure plates. Conduction heat transfer occurs

through both the water cavities and the channels. While the water jacket channels are sufficiently large for natural convection loops to form, this mechanism is conservatively neglected. Heat is passively rejected to the ambient from the outer surface of the HI-TRAC transfer overpack by natural convection and thermal radiation.

In the vertical position, the bottom face of the HI-TRAC is in contact with a supporting surface. This face is conservatively modeled as an insulated surface. Because the HI-TRAC is not used for long-term storage in an array, radiative blocking does not need to be considered. The HI-TRAC top lid is modeled as a surface with convection, radiative heat exchange with air and a constant maximum incident solar heat flux load. Insolation on cylindrical surfaces is conservatively based on 12-hour levels prescribed in 10CFR71 averaged on a 24-hour basis. Concise descriptions of these models are given below.

#### 4.5.1.1.1 Effective Thermal Conductivity of Water Jacket

The 125-ton HI-TRAC water jacket is composed of fourteen formed channels equispaced along the circumference of the HI-TRAC and welded along their length to the HI-TRAC outer shell. Enclosure plates are welded to these channels, creating twenty-eight water compartments. The 100-ton HI-TRAC water jacket has 15 formed channels and enclosure plates creating thirty compartments. Holes in the channel legs connect all the individual compartments in the water jacket. Thus, the annular region between the HI-TRAC outer shell and the enclosure shell can be considered as an array of steel ribs and water spaces.

The effective radial thermal conductivity of this array of steel ribs and water spaces is determined by combining the heat transfer resistance of individual components in a parallel network. A bounding calculation is assured by using the minimum number of channels and channel thickness as input values. The thermal conductivity of the parallel steel ribs and water spaces is given by the following formula:

$$K_{ne} = \frac{K_r N_r t_r \ln\left(\frac{r_o}{r_i}\right)}{2\pi L_R} + \frac{K_w N_r t_w \ln\left(\frac{r_o}{r_i}\right)}{2\pi L_R}$$

where:

$K_{ne}$  = effective radial thermal conductivity of water jacket

$r_i$  = inner radius of water spaces

$r_o$  = outer radius of water spaces

$K_r$  = thermal conductivity of carbon steel ribs

$N_r$  = minimum number of channel legs (equal to number of water spaces)

$t_r$  = minimum (nominal) rib thickness (lower of 125-ton and 100-ton designs)

$L_R$  = effective radial heat transport length through water spaces

$K_w$  = thermal conductivity of water

$t_w$  = water space width (between two carbon steel ribs)

Figure 4.5.1 depicts the resistance network to combine the resistances to determine an effective conductivity of the water jacket. The effective thermal conductivity is computed in the manner of the foregoing, and is provided in Table 4.5.1.

#### 4.5.1.1.2 Heat Rejection from Overpack Exterior Surfaces

The following relationship for the surface heat flux from the outer surface of an isolated cask to the environment *is* applied to the thermal model:

$$q_s = 0.19(T_s - T_A)^{4/3} + 0.1714\epsilon \left[ \left( \frac{T_s + 460}{100} \right)^4 - \left( \frac{T_A + 460}{100} \right)^4 \right]$$

where:

- $T_s$  = cask surface temperatures (°F)
- $T_A$  = ambient atmospheric temperature (°F)
- $q_s$  = surface heat flux (Btu/ft<sup>2</sup>×hr)
- $\epsilon$  = surface emissivity

The second term in this equation is the Stefan-Boltzmann formula for thermal radiation from an exposed surface to ambient. The first term is the natural convection heat transfer correlation recommended by Jacob and Hawkins [4.2.9]. This correlation is appropriate for turbulent natural convection from vertical surfaces, such as the vertical overpack wall. Although the ambient air is conservatively assumed to be quiescent, the natural convection is nevertheless turbulent.

Turbulent natural convection correlations are suitable for use when the product of the Grashof and Prandtl ( $Gr \times Pr$ ) numbers exceeds  $10^9$ . This product can be expressed as  $L^3 \times \Delta T \times Z$ , where  $L$  is the characteristic length,  $\Delta T$  is the surface-to-ambient temperature difference, and  $Z$  is a function of the surface temperature. The characteristic length of a vertically oriented HI-TRAC is its height of approximately 17 feet. The value of  $Z$ , conservatively taken at a surface temperature of 340°F, is  $2.6 \times 10^5$ . Solving for the value of  $\Delta T$  that satisfies the equivalence  $L^3 \times \Delta T \times Z = 10^9$  yields  $\Delta T = 0.78^\circ\text{F}$ . For a horizontally oriented HI-TRAC the characteristic length is the diameter of approximately 7.6 feet (minimum of 100- and 125-ton designs), yielding  $\Delta T = 8.76^\circ\text{F}$ . The natural convection will be turbulent, therefore, provided the surface to air temperature difference is greater than or equal to  $0.78^\circ\text{F}$  for a vertical orientation and  $8.76^\circ\text{F}$  for a horizontal orientation.

#### 4.5.1.1.3 Determination of Solar Heat Input

As discussed in Section 4.4.1.1.8, the intensity of solar radiation incident on an exposed surface depends on a number of time varying terms. A twelve-hour averaged insolation level is prescribed in 10CFR71 for curved surfaces. The HI-TRAC cask, however, possesses a considerable thermal inertia. This large thermal inertia precludes the HI-TRAC from reaching a steady-state thermal condition during a twelve-hour period. Thus, it is considered appropriate to use the 24-hour averaged insolation level.

#### 4.5.1.1.4 MPC Temperatures During Moisture Removal Operations

##### 4.5.1.1.4.1 Vacuum Drying

The initial loading of SNF in the MPC requires that the water within the MPC be drained and replaced with helium. For MPCs containing moderate burnup fuel assemblies only, this operation may be carried out using the conventional vacuum drying approach. In this method, removal of the last traces of residual moisture from the MPC cavity is accomplished by evacuating the MPC for a short time after draining the MPC. ~~As stipulated in the Technical Specifications, vacuum drying may not be performed on MPCs containing high burnup fuel assemblies.~~ High burnup fuel drying is performed by a forced flow helium drying process as described in Section 4.5.1.1.4.2 and Appendix 2.B.

Prior to the start of the MPC 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 active fuel length is uncovered during the draining operation, the fuel and basket mass will undergo a gradual heat up from the initially cold conditions when the heated surfaces were submerged under water.

The vacuum condition effective fuel assembly conductivity is determined by procedures discussed earlier (Subsection 4.4.1.1.2) after setting the thermal conductivity of the gaseous medium to a small fraction (one part in one thousand) of helium conductivity. The MPC basket cross sectional effective conductivity is determined for vacuum conditions according to the procedure discussed in 4.4.1.1.4. Basket periphery-to-MPC shell heat transfer occurs through conduction and radiation.

For total decay heat loads up to and including 20.88 kW for the MPC-24 and 21.52 kW for the MPC-68, vacuum drying of the MPC is performed with the annular gap between the MPC and the HI-TRAC filled with water. The presence of water in this annular gap will maintain the MPC shell temperature approximately equal to the saturation temperature of the annulus water. Thus, the thermal analysis of the MPC during vacuum drying for these conditions is performed with cooling of the MPC shell with water at a bounding maximum temperature of 232°F.

For higher total decay heat loads in the MPC-24 and MPC-68 or for any decay heat load in an MPC-24E or MPC-32, vacuum drying of the MPC is performed with the annular gap between the MPC and the HI-TRAC continuously flushed with water. The water movement in this annular gap will maintain the MPC shell temperature at about the temperature of flowing water. Thus, the thermal analysis of the MPC during vacuum drying for these conditions is performed with cooling of the MPC shell with water at a bounding maximum temperature of 125°F.

An axisymmetric FLUENT thermal model of the MPC is constructed, employing the MPC in-plane conductivity as an isotropic fuel basket conductivity (i.e. conductivity in the basket radial and axial directions is equal), to determine peak cladding temperature at design basis heat loads.

To avoid excessive conservatism in the computed FLUENT solution, partial recognition for higher axial heat dissipation is adopted in the peak cladding calculations. The boundary conditions applied to this evaluation are:

- i. A bounding steady-state analysis is performed with the MPC decay heat load set equal to the largest design-basis decay heat load. As discussed above, there are two different ranges for the MPC-24 and MPC-68 designs.
- ii. The entire outer surface of the MPC shell is postulated to be at a bounding maximum temperature of 232°F or 125°F, as discussed above.
- iii. The top and bottom surfaces of the MPC are adiabatic.

Results of vacuum condition analyses are provided in Subsection 4.5.2.2.

#### 4.5.1.1.4.2 Forced Helium Dehydration Recirculation

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a conventional, closed loop dehumidification system consisting of a condenser, a demister, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulence. A vapor pressure of 3 torr or less is assured by verifying that the helium temperature exiting the demister is maintained at or below the psychrometric threshold of 21°F for a minimum of 30 minutes. See Appendix 2.B for detailed discussion of the design criteria and operation of the FHD system.

The FHD system provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHD system ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit for normal conditions of storage, which is well below the high burnup cladding temperature limit 752°F (400°C) for all combinations of SNF type, burnup, decay heat, and cooling time. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode of natural convection in long term storage), it is readily concluded that the peak fuel cladding temperature under the latter condition will be greater than that during the FHD operation phase. In the event that the FHD system malfunctions, the forced convection state will degenerate to natural convection, which corresponds to the conditions of normal *onsite transport storage*. As a result, the peak fuel cladding temperatures will approximate the values reached during normal *onsite transport storage* as described elsewhere in this chapter.

#### 4.5.1.1.5 Maximum Time Limit During Wet Transfer Operations

In accordance with NUREG-1536, water inside the MPC cavity during wet transfer operations is not permitted to boil. Consequently, uncontrolled pressures in the de-watering, purging, and

recharging system that may result from two-phase conditions are completely avoided. This requirement is accomplished by imposing a limit on the maximum allowable time duration for fuel to be submerged in water after a loaded HI-TRAC cask is removed from the pool and prior to the start of vacuum drying operations.

When the HI-TRAC transfer cask and the loaded MPC under water-flooded conditions are removed from the pool, the combined water, fuel mass, MPC, and HI-TRAC metal will absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the entire system with time, starting from an initial temperature of the contents. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC system. To enable a bounding heat-up rate determination for the HI-TRAC system, the following conservative assumptions are imposed:

- i. Heat loss by natural convection and radiation from the exposed HI-TRAC surfaces to the pool building ambient air is neglected (i.e., an adiabatic temperature rise calculation is performed).
- ii. Design-basis maximum decay heat input from the loaded fuel assemblies is imposed on the HI-TRAC transfer cask.
- iii. The smaller of the two (i.e., 100-ton and 125-ton) HI-TRAC transfer cask designs 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.
- iv. The smallest of the minimum MPC cavity-free volumes among the two MPC types is considered for flooded water mass determination.
- v. Only fifty percent of the water mass in the MPC cavity is credited towards water thermal inertia evaluation.

Table 4.5.5 summarizes the weights and thermal inertias of several 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 governed by the following equation:

$$\frac{dT}{dt} = \frac{Q}{C_h}$$

where:

- Q = decay heat load (Btu/hr) [Design Basis maximum 28.74 kW = 98,205 Btu/hr]  
C<sub>h</sub> = combined thermal inertia of the loaded HI-TRAC transfer cask (Btu/°F)  
T = temperature of the contents (°F)  
t = time after HI-TRAC transfer cask is removed from the pool (hr)



A bounding heat-up rate for the HI-TRAC transfer cask contents is determined to be equal to 3.77°F/hr. From this adiabatic rate of temperature rise estimate, the maximum allowable time duration ( $t_{\max}$ ) for fuel to be submerged in water is determined as follows:

$$t_{\max} = \frac{T_{\text{boil}} - T_{\text{initial}}}{(dT/dt)}$$

where:

$T_{\text{boil}}$  = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)

$T_{\text{initial}}$  = initial temperature of the HI-TRAC contents when the transfer cask is removed from the pool

Table 4.5.6 provides a summary of  $t_{\max}$  at several representative HI-TRAC contents starting temperature.

As set forth in the HI-STORM operating procedures, in the unlikely event that the maximum allowable time provided in Table 4.5.6 is found to be insufficient to complete all wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water will enter via the MPC lid drain port connection and heated water will exit from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

$$M_w = \frac{Q}{C_{pw} (T_{\max} - T_{in})}$$

where:

$M_w$  = minimum water flow rate (lb/hr)

$C_{pw}$  = water heat capacity (Btu/lb-°F)

$T_{\max}$  = maximum MPC cavity water mass temperature

$T_{in}$  = temperature of pool water supply to MPC

With the MPC cavity water temperature limited to 150°F, MPC inlet water maximum temperature equal to 125°F and at the design basis maximum heat load, the water flow rate is determined to be 3928 lb/hr (7.9 gpm).

#### 4.5.1.1.6 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

NUREG-1536 requires an evaluation of cask cooldown and reflood procedures to support fuel unloading from a dry condition. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. The extremely rapid cooldown rates to which the hot MPC internals and the fuel cladding are subjected during water injection may, however, result in uncontrolled thermal stresses and failure in the structural members. Moreover, water injection results in large amounts of steam generation and unpredictable transient two-phase flow conditions inside the MPC cavity, which may result in

overpressurization of the confinement boundary. To avoid potential safety concerns related to rapid cask cooldown by direct water quenching, the HI-STORM MPCs will be cooled in a gradual manner, thereby eliminating thermal shock loads on the MPC internals and fuel cladding.

In the unlikely event that a HI-STORM storage system is required to be unloaded, the MPC will be transported on-site via the HI-TRAC transfer cask back to the fuel handling building. Prior to reflooding the MPC cavity with water<sup>†</sup>, a forced flow helium recirculation system with adequate flow capacity shall be operated to remove the decay heat and initiate a slow cask cooldown lasting for several days. The operating procedures in Chapter 8 (Section 8.3) provide a detailed description of the steps involved in the cask unloading. An analytical method that provides a basis for determining the required helium flow rate as a function of the desired cooldown time is presented below, to meet the objective of eliminating thermal shock when the MPC cavity is eventually flooded with water.

Under a closed-loop forced helium circulation condition, the helium gas is cooled, via an external chiller, down to 100°F. The chilled helium is then introduced into the MPC cavity, near the MPC baseplate, through the drain line. The helium gas enters the MPC basket from the bottom oversized flow holes and moves upward through the hot fuel assemblies, removing heat and cooling the MPC internals. The heated helium gas exits from the top of the basket and collects in the top plenum, from where it is expelled through the MPC lid vent connection to the helium recirculation and cooling system. The MPC contents bulk average temperature reduction as a function of time is principally dependent upon the rate of helium circulation. The temperature transient is governed by the following heat balance equation:

$$C_h \frac{dT}{dt} = Q_D - m C_p (T - T_i) - Q_c$$

Initial Condition:  $T = T_o$  at  $t = 0$

where:

- $T =$  MPC bulk average temperature (°F)
- $T_o =$  initial MPC bulk average temperature in the HI-TRAC transfer cask (equal to 586°F)
- $t =$  time after start of forced circulation (hrs)
- $Q_D =$  decay heat load (Btu/hr)  
(equal to Design Basis maximum 28.74kW (i.e., 98,205 Btu/hr))
- $m =$  helium circulation rate (lb/hr)
- $C_p =$  helium heat capacity (Btu/lb-°F)  
(equal to 1.24 Btu/lb-°F)

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<sup>†</sup> Prior to helium circulation, the HI-TRAC annulus is flooded with water to substantially lower the MPC shell temperature (approximately 100°F). For low decay heat MPCs (~10 kW or less) the annulus cooling is adequate to lower the MPC cavity temperature below the boiling temperature of water.

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- $Q_c$  = heat rejection from cask exposed surfaces to ambient (Btu/hr) (conservatively neglected)
- $C_h$  = thermal capacity of the loaded MPC (Btu/°F)  
(For a bounding upper bound 100,000 lb loaded MPC weight and heat capacity of Alloy X equal to 0.12 Btu/lb-°F, the heat capacity is equal to 12,000 Btu/°F.)
- $T_i$  = MPC helium inlet temperature (°F)

The differential equation is analytically solved, yielding the following expression for time-dependent MPC bulk temperature:

$$T(t) = \left(T_i + \frac{Q_D}{m C_p}\right) \left(1 - e^{-\frac{m C_p}{C_h} t}\right) + T_o e^{-\frac{m C_p}{C_h} t}$$

This equation is used to determine the minimum helium mass flow rate that would cool the MPC cavity down from initially hot conditions to less than 200°F (i.e., with a subcooling margin for normal boiling temperature of water† (212°F)). For example, to cool the MPC to less than 200°F in 72 hours using 0°F helium would require a helium mass flow rate of 432 lb/hr (i.e., 647 SCFM).

Once the helium gas circulation has cooled the MPC internals to less than 200°F, water can be injected to the MPC without risk of boiling and the associated thermal stress concerns. Because of the relatively long cooldown period, the thermal stress contribution to the total cladding stress would be negligible, and the total stress would therefore be bounded by the normal (dry) condition. The elimination of boiling eliminates any concern of overpressurization due to steam production.

#### 4.5.1.1.7 Study of Lead-to-Steel Gaps on Predicted Temperatures

Lead, poured between the inner and outer shells, is utilized as a gamma shield material in the HI-TRAC on-site transfer cask designs. Lead shrinks during solidification requiring the specification and implementation of appropriate steps in the lead installation process so that the annular space is free of gaps. Fortunately, the lead pouring process is a mature technology and proven methods to insure that radial gaps do not develop are widely available. This subsection outlines such a method to achieve a zero-gap lead installation in the annular cavity of the HI-TRAC casks.

The 100-ton and 125-ton HI-TRAC designs incorporate 2.5 inch and 4.5 inch annular spaces, respectively, formed between a 3/4-inch thick steel inner shell and a 1-inch thick steel outer shell. The interior steel surfaces are cleaned, sandblasted and fluxed in preparation for the molten lead that will be poured in the annular cavity. The appropriate surface preparation technique is essential to ensure that molten lead sticks to the steel surfaces, which will form a metal to lead bond upon solidification. The molten lead is poured to fill the annular cavity. The

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† Certain fuel configurations in PWR MPCs are required to be flooded with borated water, which has a higher boiling temperature. Thus, greater subcooling margins are present in this case.

molten lead in the immediate vicinity of the steel surfaces, upon cooling by the inner and outer shells, solidifies forming a melt-solid interface. The initial formation of a gap-free interfacial bond between the solidified lead and steel surfaces initiates a process of lead crystallization from the molten pool onto the solid surfaces. Static pressure from the column of molten lead further aids in retaining the solidified lead layer to the steel surfaces. The melt-solid interface growth occurs by freezing of successive layers of molten lead as the heat of fusion is dissipated by the solidified metal and steel structure enclosing it. This growth stops when all the molten lead is used up and the annulus is filled with a solid lead plug. The shop fabrication procedures, being developed in conjunction with the designated manufacturer of the HI-TRAC transfer casks, shall contain detailed step-by-step instructions devised to eliminate the incidence of annular gaps in the lead space of the HI-TRAC.

In the spirit of a defense-in-depth approach, however, a conservatively bounding lead-to-steel gap is assumed herein and the resultant peak cladding temperature under design basis heat load is computed. It is noted that in a non-bonding lead pour scenario, the lead shrinkage resulting from phase transformation related density changes introduces a tendency to form small gaps. This tendency is counteracted by gravity induced slump, which tends to push the heavy mass of lead against the steel surfaces. If the annular molten mass of lead is assumed to contract as a solid, in the absence of gravity, then a bounding lead-to-steel gap is readily computed from density changes. This calculation is performed for the 125-ton HI-TRAC transfer cask, which has a larger volume of lead and is thus subject to larger volume shrinkage relative to the 100-ton design, and is presented below.

The densities of molten ( $\rho_l$ ) and solid ( $\rho_s$ ) lead are given on page 3-96 of Perry's Handbook (6<sup>th</sup> Edition) as 10,430 kg/m<sup>3</sup> and 11,010 kg/m<sup>3</sup>, respectively. The fractional volume contraction during solidification ( $\delta v/v$ ) is calculated as:

$$\frac{\delta v}{v} = \frac{(\rho_s - \rho_l)}{\rho_l} = \frac{(11,010 - 10,430)}{10,430} = 0.0556$$

and the corresponding fractional linear contraction during solidification is calculated as:

$$\frac{\delta L}{L} = \left[ 1 + \frac{\delta v}{v} \right]^{1/3} - 1 = 1.0556^{1/3} - 1 = 0.0182$$

The bounding lead-to-steel gap, which is assumed filled with air, is calculated by multiplying the nominal annulus radial dimension (4.5 inches in the 125-ton HI-TRAC) by the fractional linear contraction as:

$$\delta = 4.5 \times \frac{\delta L}{L} = 4.5 \times 0.0182 = 0.082 \cdot \text{inches}$$

In this hypothetical lead shrinkage process, the annular lead cylinder will contract towards the inner steel shell, eliminating gaps and tightly compressing the two surfaces together. Near the

outer steel cylinder, a steel-to-lead air gap will develop as a result of volume reduction in the liquid to solid phase transformation. The air gap is conservatively postulated to occur between the inner steel shell and the lead, where the heat flux is higher relative to the outer steel shell, and hence the computed temperature gradient is greater. The combined resistance of an annular lead cylinder with an air gap ( $R_{cyl}$ ) is computed by the following formula:

$$R_{cyl} = \frac{\ln(R_o/R_i)}{2\pi K_{pb}} + \frac{\delta}{2\pi R_i [K_{air} + K_r]}$$

where:

- $R_i$  = inner radius (equal to 35.125 inches)
- $R_o$  = outer radius (equal to 39.625 inches)
- $K_{pb}$  = bounding minimum lead conductivity (equal to 16.9 Btu/ft-hr-°F, from Table 4.2.2)
- $\delta$  = lead-to-steel air gap, computed above
- $K_{air}$  = temperature dependent air conductivity (see Table 4.2.2)
- $K_r$  = effective thermal conductivity contribution from radiation heat transfer across air gap

The effective thermal conductivity contribution from radiation heat transfer ( $K_r$ ) is defined by the following equation:

$$K_r = 4 \times \sigma \times F_\epsilon \times T^3 \times \delta$$

where:

- $\sigma$  = Stefan-Boltzmann constant
- $F_\epsilon = (1/\epsilon_{cs} + 1/\epsilon_{pb} - 1)^{-1}$
- $\epsilon_{cs}$  = carbon steel emissivity (equal to 0.66, HI-STORM FSAR Table 4.2.4)
- $\epsilon_{pb}$  = lead emissivity (equal to 0.63 for oxidized surfaces at 300°F from McAdams, Heat Transmission, 3<sup>rd</sup> Ed.)
- $T$  = absolute temperature

Based on the total annular region resistance ( $R_{cyl}$ ) computed above, an equivalent annulus conductivity is readily computed. This effective temperature-dependent conductivity results are tabulated below:

Temperature (°F)	Effective Annulus Conductivity (Btu/ft-hr-°F)
200	1.142
450	1.809

The results tabulated above confirm that the assumption of a bounding annular air gap grossly penalizes the heat dissipation characteristics of lead filled regions. Indeed, the effective conductivity computed above is an order of magnitude lower than that of the base lead material. To confirm the heat dissipation adequacy of HI-TRAC casks under the assumed overly

pessimistic annular gaps, the HI-TRAC thermal model described earlier is altered to include the effective annulus conductivity computed above for the annular lead region. The peak cladding temperature results are tabulated below:

<b>Annular Gap Assumption</b>	<b>Peak Cladding Temperature (°F)</b>	<b>Cladding Temperature Limit (°F)</b>
None	872	1058
Bounding Maximum	924	1058

From these results, it is readily apparent that the stored fuel shall be maintained within safe temperature limits by a substantial margin of safety (in excess of 100°F).

#### 4.5.1.2 Test Model

A detailed analytical model for thermal design of the HI-TRAC transfer cask was developed using the FLUENT CFD code, the industry standard ANSYS modeling package and conservative adiabatic calculations, as discussed in Subsection 4.5.1.1. Furthermore, the analyses incorporate many conservative assumptions in order to demonstrate compliance to the specified short-term limits with adequate margins. In view of these considerations, the HI-TRAC transfer cask thermal design complies with the thermal criteria established for short-term handling and onsite transport. Additional experimental verification of the thermal design is therefore not required.

#### 4.5.2 Maximum Temperatures

##### 4.5.2.1 Maximum Temperatures Under Onsite Transport Conditions

An axisymmetric FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was developed to evaluate temperature distributions for onsite transport conditions. A bounding steady-state analysis of the HI-TRAC transfer cask has been performed using the hottest MPC, the highest design-basis decay heat load (Table 2.1.6), and design-basis insolation levels. While the duration of onsite transport may be short enough to preclude the MPC and HI-TRAC from obtaining a steady-state, a steady-state analysis is conservative. Information listing all other thermal analyses pertaining to the HI-TRAC cask and associated subsection of the FSAR summarizing obtained results is provided in Table 4.5.8.

A converged temperature contour plot is provided in Figure 4.5.2. Maximum fuel clad temperatures are listed in Table 4.5.2, which also summarizes maximum calculated temperatures in different parts of the HI-TRAC transfer cask and MPC. As described in Subsection 4.4.2, the FLUENT calculated peak temperature in Table 4.5.2 is actually the peak pellet centerline temperature, which bounds the peak cladding temperature. We conservatively assume that the peak clad temperature is equal to the peak pellet centerline temperature.

The maximum computed temperatures listed in Table 4.5.2 are based on the HI-TRAC cask at Design Basis Maximum heat load, passively rejecting heat by natural convection and radiation to a hot ambient environment at 100°F in still air in a vertical orientation. In this orientation, there is apt to be a less of metal-to-metal contact between the physically distinct entities, viz., fuel, fuel basket, MPC shell and HI-TRAC cask. For this reason, the gaps resistance between these parts is higher than in a horizontally oriented HI-TRAC. To bound gaps resistance, the various parts are postulated to be in a centered configuration. MPC internal convection at a postulated low cavity pressure of 5 atm is included in the thermal model. The peak cladding temperature computed under these adverse Ultimate Heat Sink (UHS) assumptions is 872°F which is substantially lower than the ~~short-term~~ temperature limit of 1058°F *for moderate burnup fuel (MBF)*. Consequently, cladding integrity assurance is provided by large safety margins (in excess of 100°F) during onsite transfer of an MPC *containing MBF* emplaced in a HI-TRAC cask.

As a defense-in-depth measure, cladding integrity is demonstrated for a theoretical bounding scenario. For this scenario, all means of convective heat dissipation within the canister are neglected in addition to the bounding relative configuration for the fuel, basket, MPC shell and HI-TRAC overpack assumption stated earlier for the vertical orientation. This means that the fuel is centered in the basket cells, the basket is centered in the MPC shell and the MPC shell is centered in the HI-TRAC overpack to maximize gaps thermal resistance. The peak cladding temperature computed for this scenario (1025°F) is below the short-term limit of 1058°F.

*For high burnup fuel (HBF), however, the maximum computed fuel cladding temperature reported in Table 4.5.2 is significantly greater than the temperature limit of 752°F for HBF. Consequently, it is necessary to utilize the SCS described at the beginning of this section and in Appendix 2.C during onsite transfer of an MPC containing HBF emplaced in a HI-TRAC transfer cask. As stated earlier, the exact design and operation of the SCS is necessarily site-specific. The design is required to satisfy the specifications and operational requirements of Appendix 2.C to ensure compliance with ISG-11 [4.1.4] temperature limits.*

As discussed in Sub-section 4.5.1.1.6, MPC fuel unloading operations are performed with the MPC inside the HI-TRAC cask. For this operation, a helium cooldown system is engaged to the MPC via lid access ports and a forced helium cooling of the fuel and MPC is initiated. With the HI-TRAC cask external surfaces dissipating heat to a UHS in a manner in which the ambient air access is not restricted by bounding surfaces or large objects in the immediate vicinity of the cask, the temperatures reported in Table 4.5.2 will remain bounding during fuel unloading operations. ~~Under a scenario in which the cask is emplaced in a area with ambient air access restrictions (for example in a cask pit area), additional means shall be devised to limit the cladding temperature rise arising from such restrictions to less than 100°F. These means are discussed next.~~

~~The time duration allowed for the cask to be emplaced in a ambient air restricted area with the helium cooling system non-operational shall be limited to 22 hours. Conservatively postulating that the rate of passive cooling is substantially degraded by 90% (i.e., 10% of decay heat is~~

dissipated to ambient), cladding integrity is demonstrated based on cask heating considerations from the undissipated heat. At a bounding heat load of 28.74kW, the HI-TRAC cask system thermal inertia (19,532 Btu/°F, Table 4.5.5), limits the temperature rise to 4.52°F/hr. Thus, the computed cladding temperature rise during this time period will be less than 100°F.

A forced supply of ambient air near the bottom of the cask pit to aid heat dissipation by the natural convection process is another adequate means to maintain the fuel cladding within safe operating limits. Conservatively assuming this column of moving air as the UHS (i.e. to which all heat dissipation occurs) with no credit for enhanced cooling as a result of forced convection heat transfer, a nominal air supply of 1000 SCFM (4850 lbs/hr) adequately meets the cooling requirement. At this flow rate, the temperature rise of the UHS resulting from cask decay heat input to the airflow will be less than 100°F. The cladding temperature elevation will consequently be bounded by this temperature rise.

#### 4.5.2.2 Maximum MPC Basket Temperature Under Vacuum Conditions

As stated in Subsection 4.5.1.1.4, above, an axisymmetric FLUENT thermal model of the MPC is developed for the vacuum condition. For the MPC-24E and MPC-32 designs, and for the higher heat load ranges in the MPC-24 and MPC-68 designs, the model also includes an isotropic fuel basket thermal conductivity. Each MPC is analyzed at its respective design maximum heat load. The steady-state peak cladding results, with partial recognition for higher axial heat dissipation where included, are summarized in Table 4.5.9. The peak fuel clad temperatures for moderate burnup fuel during short-term vacuum drying operations with design-basis maximum heat loads are calculated to be less than 1058°F for all MPC baskets by a significant margin.

#### 4.5.3 Minimum Temperatures

In Table 2.2.2 and Chapter 12, the minimum ambient temperature condition required to be considered for the HI-TRAC design is specified as 0°F. If, conservatively, a zero decay heat load (with no solar input) is applied to the stored fuel assemblies then every component of the system at steady state would be at this outside minimum temperature. Provided an antifreeze is added to the water jacket (required by Technical Specification for ambient temperatures below 32°F), all HI-TRAC materials will satisfactorily perform their intended functions at this minimum postulated temperature condition. Fuel transfer operations are ~~must be~~ controlled by ~~Technical Specifications in Chapter 12~~ to ensure that onsite transport operations are not performed at an ambient temperature less than 0°F.

#### 4.5.4 Maximum Internal Pressure

After fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling in the HI-TRAC transfer cask, the gas temperature within the MPC rises to its maximum operating temperature as determined based on the thermal



analysis methodology described previously. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined based on the ideal gas law, which states that the absolute pressure of a fixed volume of gas is proportional to its absolute temperature. The net free volumes of the four MPC designs are determined in Section 4.4.

The maximum MPC internal pressure is determined for normal onsite transport conditions, as well as off-normal conditions of a postulated accidental release of fission product gases caused by fuel rod rupture. Based on NUREG-1536 [4.4.10] recommended fission gases release fraction data, net free volume and initial fill gas pressure, the bounding maximum gas pressures with 1% and 10% rod rupture are given in Table 4.5.3. The MPC maximum gas pressures listed in Table 4.5.3 are all below the MPC design internal pressure listed in Table 2.2.1.

#### 4.5.5 Maximum Thermal Stresses

Thermal expansion induced mechanical stresses due to non-uniform temperature distributions are reported in Chapter 3. Tables 4.5.2 and 4.5.4 provide a summary of MPC and HI-TRAC transfer cask component temperatures for structural evaluation.

#### 4.5.6 Evaluation of System Performance for Normal Conditions of Handling and Onsite Transport

The HI-TRAC transfer cask thermal analysis is based on a detailed heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and HI-TRAC. The thermal model incorporates several conservative features, which are listed below:

- i. The most severe levels of environmental factors - bounding ambient temperature (100°F) and constant solar flux - were coincidentally imposed on the thermal design. A bounding solar absorptivity of 1.0 is applied to all insolation surfaces.
- ii. The HI-TRAC cask-to-MPC annular gap is analyzed based on the nominal design dimensions. No credit is considered for the significant reduction in this radial gap that would occur as a result of differential thermal expansion with design basis fuel at hot conditions. The MPC is considered to be concentrically aligned with the cask cavity. This is a worst-case scenario since any eccentricity will improve conductive heat transport in this region.
- iii. No credit is considered for cooling of the HI-TRAC baseplate while in contact with a supporting surface. An insulated boundary condition is applied in the thermal model on the bottom baseplate face.

Temperature distribution results (Tables 4.5.2 and 4.5.4, and Figure 4.5.2) obtained from this highly conservative thermal model show that the ~~short-term~~ fuel cladding and cask component temperature limits are met with adequate margins *for MBF. For HBF, supplemental cooling is required to comply with the applicable temperature limits.* Expected margins during normal HI-

TRAC use will be larger due to the many conservative assumptions incorporated in the analysis. Corresponding MPC internal pressure results (Table 4.5.3) show that the MPC confinement boundary remains well below the short-term condition design pressure. Stresses induced due to imposed temperature gradients are within ASME Code limits (Chapter 3). The maximum local axial neutron shield temperature is lower than design limits. Therefore, it is concluded that the HI-TRAC transfer cask thermal design is adequate to maintain fuel cladding integrity for short-term onsite handling and transfer operations.

The water in the water jacket of the HI-TRAC provides necessary neutron shielding. During normal handling and onsite transfer operations this shielding water is contained within the water jacket, which is designed for an elevated internal pressure. It is recalled that the water jacket is equipped with pressure relief valves set at 60 psig and 65 psig. This set pressure elevates the saturation pressure and temperature inside the water jacket, thereby precluding boiling in the water jacket under normal conditions. Under normal handling and onsite transfer operations, the bulk temperature inside the water jacket reported in Table 4.5.2 is less than the coincident saturation temperature at 60 psig (307°F), so the shielding water remains in its liquid state. The bulk temperature is determined via a conservative analysis, presented earlier, with design-basis maximum decay heat load. One of the assumptions that render the computed temperatures extremely conservative is the stipulation of a 100°F steady-state ambient temperature. In view of the large thermal inertia of the HI-TRAC, an appropriate ambient temperature is the “time-averaged” temperature, formally referred to in this FSAR as the normal temperature.

Note that during hypothetical fire accident conditions (see Section 11.2) these relief valves allow venting of any steam generated by the extreme fire flux, to prevent overpressurizing the water jacket. In this manner, a portion of the fire heat flux input to the HI-TRAC outer surfaces is expended in vaporizing a portion of the water in the water jacket, thereby mitigating the magnitude of the heat input to the MPC during the fire.

During vacuum drying operations, the annular gap between the MPC and the HI-TRAC is filled with water. The saturation temperature of the annulus water bounds the maximum temperatures of all HI-TRAC components, which are located radially outside the water-filled annulus. As previously stated (see Subsection 4.5.1.1.4) the maximum annulus water temperature is only 125°F, so the HI-TRAC water jacket temperature will be less than the 307°F saturation temperature.

Table 4.5.1

EFFECTIVE RADIAL THERMAL CONDUCTIVITY OF THE WATER JACKET

<b>Temperature (°F)</b>	<b>Thermal Conductivity (Btu/ft-hr-°F)</b>
200	1.376
450	1.408
700	1.411

Table 4.5.2

HI-TRAC TRANSFER CASK STEADY-STATE  
MAXIMUM TEMPERATURES

Component	Temperature [°F]
Fuel Cladding	872*
MPC Basket	852
Basket Periphery	600
MPC Outer Shell Surface	455
HI-TRAC Overpack Inner Surface	322
Water Jacket Inner Surface	314
Enclosure Shell Outer Surface	224
Water Jacket Bulk Water	258
Axial Neutron Shield <sup>†</sup>	258

\* *This calculated value exceeds the allowable limit for high-burnup fuel. A Supplemental Cooling System that satisfies the criteria in Appendix 2.C shall be used to comply with applicable temperature limits when an MPC contains one or more high burnup fuel assemblies.*

<sup>†</sup> Local neutron shield section temperature.

Table 4.5.3

SUMMARY OF MPC CONFINEMENT BOUNDARY PRESSURES<sup>†</sup> FOR  
NORMAL HANDLING AND ONSITE TRANSPORT

Condition	Pressure (psig)
MPC-24:	
Initial backfill (at 70°F)	31.3
Normal condition	76.0
With 1% rod rupture	76.8
With 10% rod rupture	83.7
MPC-68:	
Initial backfill (at 70°F)	31.3
Normal condition	76.0
With 1% rods rupture	76.5
With 10% rod rupture	80.6
MPC-32:	
Initial backfill (at 70°F)	31.3
Normal condition	76.0
With 1% rods rupture	77.1
With 10% rod rupture	86.7
MPC-24E:	
Initial backfill (at 70°F)	31.3
Normal condition	76.0
With 1% rods rupture	76.8
With 10% rod rupture	83.7

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<sup>†</sup> Includes gas from BPRA rods for PWR MPCs

Table 4.5.4

SUMMARY OF HI-TRAC TRANSFER CASK AND MPC COMPONENTS  
NORMAL HANDLING AND ONSITE TRANSPORT TEMPERATURES

Location	Temperature (°F)
MPC Basket Top: Basket periphery MPC shell O/P <sup>†</sup> inner shell O/P enclosure shell	590 445 280 196
MPC Basket Bottom: Basket periphery MPC shell O/P inner shell O/P enclosure shell	334 302 244 199

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<sup>†</sup> O/P is an abbreviation for HI-TRAC overpack.

Table 4.5.5

SUMMARY OF LOADED 100-TON HI-TRAC TRANSFER CASK  
 BOUNDING COMPONENT  
 WEIGHTS AND THERMAL INERTIAS

<b>Component</b>	<b>Weight (lbs)</b>	<b>Heat Capacity (Btu/lb-°F)</b>	<b>Thermal Inertia (Btu/°F)</b>
Water Jacket	7,000	1.0	7,000
Lead	52,000	0.031	1,612
Carbon Steel	40,000	0.1	4,000
Alloy-X MPC (empty)	39,000	0.12	4,680
Fuel	40,000	0.056	2,240
MPC Cavity Water <sup>†</sup>	6,500	1.0	6,500
			26,032 (Total)

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<sup>†</sup> Conservative lower bound water mass.

Table 4.5.6

MAXIMUM ALLOWABLE TIME DURATION FOR WET  
TRANSFER OPERATIONS

<b>Initial Temperature (°F)</b>	<b>Time Duration (hr)</b>
115	25.7
120	24.4
125	23.1
130	21.7
135	20.4
140	19.1
145	17.8
150	16.4



Table 4.5.7

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Table 4.5.8

MATRIX OF HI-TRAC TRANSFER CASK THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Onsite Transport	Ambient	SS(B)	O <sub>T</sub> , Q <sub>D</sub> , ST, SC	4.5.2.1
2	Lead Gaps	Ambient	SS(B)	O <sub>T</sub> , Q <sub>D</sub> , ST, SC	4.5.1.1.7
3	Vacuum	HI-TRAC annulus water	SS(B)	Q <sub>D</sub>	4.5.2.2
4	Wet Transfer Operation	Cavity water and Cask Internals	AH	Q <sub>D</sub>	4.5.1.1.5
5	Fuel Unloading	Helium Circulation	TA	Q <sub>D</sub>	4.5.1.1.6
6	Fire Accident	Jacket Water, Cask Internals	TA	Q <sub>D</sub> , F	11.2.4
7	Jacket Water Loss Accident	Ambient	SS(B)	O <sub>T</sub> , Q <sub>D</sub> , ST, SC	11.2.1

Legend:

O<sub>T</sub> - Off-Normal Temperature (100°F)  
 Q<sub>D</sub> - Design Basis Maximum Heat Load

SS(B) - Bounding Steady State  
 TA - Transient Analysis  
 AH - Adiabatic Heating

ST - Insolation Heating (Top)  
 SC - Insolation Heating (Curved)  
 F - Fire Heating (1475°F)

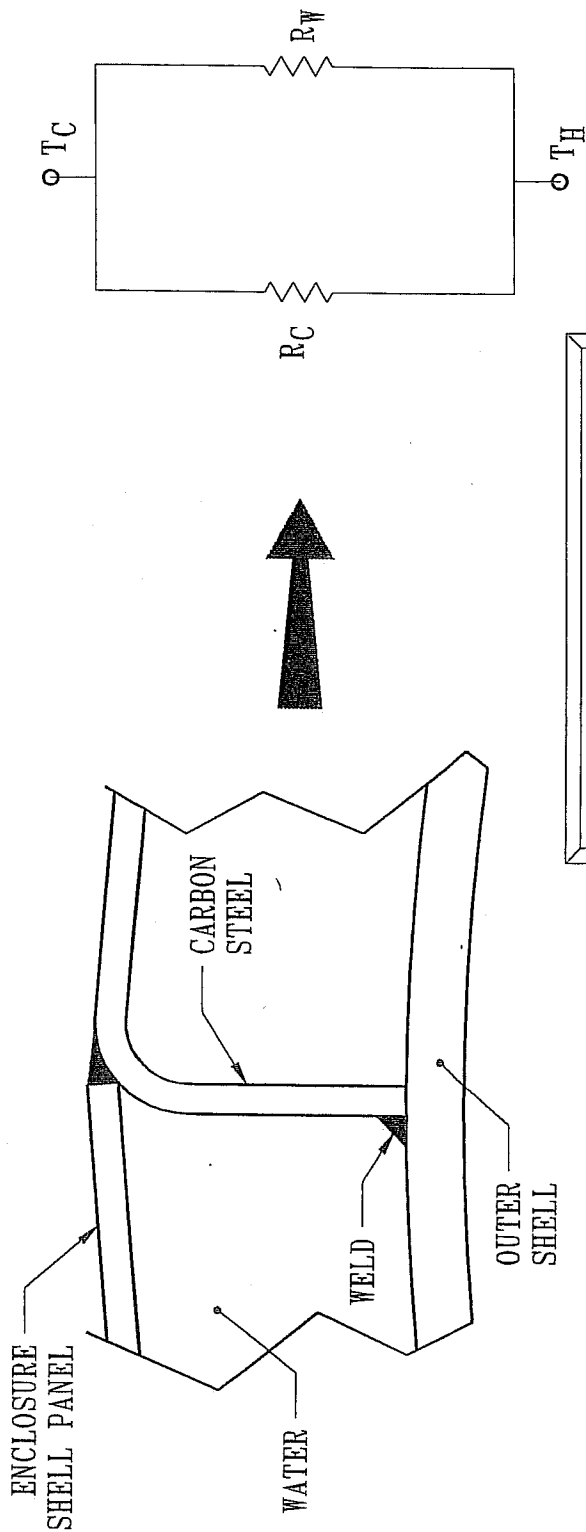
Table 4.5.9

PEAK CLADDING TEMPERATURE IN VACUUM<sup>†</sup>  
 (MODERATE BURNUP FUEL ONLY)

<b>MPC</b>	<b>Lower Decay Heat Load Range Temperatures (°F)</b>	<b>Higher Decay Heat Load Range Temperature (°F)</b>
MPC-24	827	960
MPC-68	822	1014
MPC-32	n/a	1040
MPC-24E	n/a	942

---

<sup>†</sup> Steady state temperatures at the MPC design maximum heat load reported.



$R_C$ : CARBON STEEL RESISTANCE  
 $R_W$ : WATER RESISTANCE  
 $T_H$ : HOT TEMPERATURES  
 $T_C$ : COLD TEMPERATURES

FIGURE 4.5.1; WATER JACKET RESISTANCE NETWORK ANALOGY FOR EFFECTIVE CONDUCTIVITY CALCULATION

7.40E+02  
 7.27E+02  
 7.14E+02  
 7.01E+02  
 6.88E+02  
 6.75E+02  
 6.63E+02  
 6.50E+02  
 6.37E+02  
 6.24E+02  
 6.11E+02  
 5.98E+02  
 5.85E+02  
 5.73E+02  
 5.60E+02  
 5.47E+02  
 5.34E+02  
 5.21E+02  
 5.08E+02  
 4.95E+02  
 4.83E+02  
 4.70E+02  
 4.57E+02  
 4.44E+02  
 4.31E+02  
 4.18E+02  
 4.05E+02  
 3.93E+02  
 3.80E+02  
 3.67E+02  
 3.54E+02

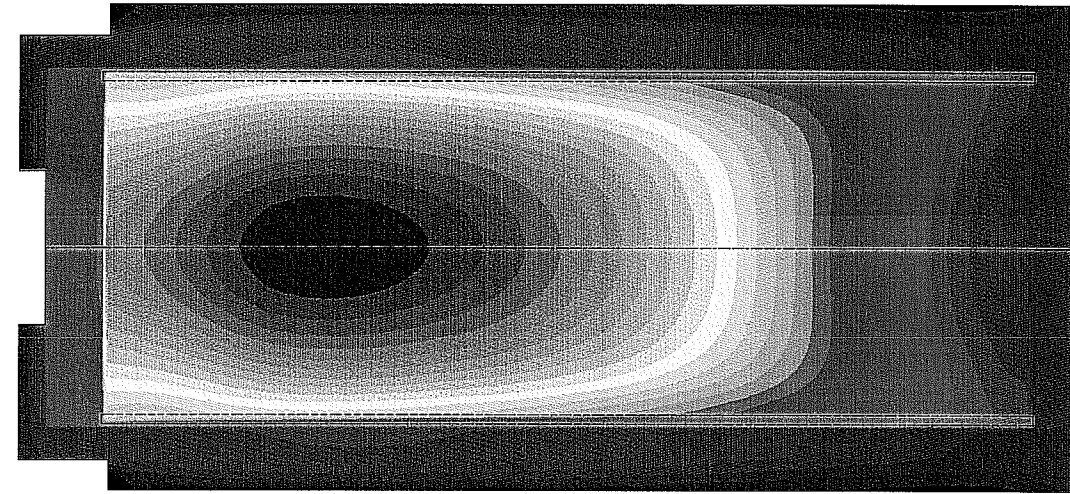


FIGURE 4.5.2: HI-TRAC Temperature Contours Plot  
 Temperature (Degrees Kelvin)  
 Max = 7.398E+02 Min = 3.540E+02

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 Fluent 4.48  
 Fluent Inc.

FIGURE 4.5.3  
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## 4.6 REGULATORY COMPLIANCE

### 4.6.1 Normal Conditions of Storage

NUREG-1536 [4.4.10] and ISG-11 [4.1.4] defines several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 4.1 through 4.4.5 and results evaluated in Subsection 4.4.6. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by ISG-11 [4.1.4] NUREG-1536 (4.0,IV,1), the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System and a minimum of 20 years of cask storage. Maximum clad temperatures for long-term storage conditions are reported in Section 4.4.2. Anticipated damage threshold temperatures, calculated as described in Section 4.3, are summarized in Table 2.2.3.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Section 4.4.4. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for long-term storage conditions are reported in Sections 4.4.2 and 4.4.3, respectively. Design temperature limits are summarized in Table 2.2.3. HI-STORM System components defined as important to safety are listed in Table 2.2.6.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 [4.1.4] limit of 400 °C (752 °F). Further, NUREG-1536 (4.0,IV,6) requires that the fuel cladding damage resulting from creep cavitation should be limited to 15 percent of the original cladding cross section area during dry storage. The calculation methodology, described in Section 4.3, for determining initial dry storage fuel clad temperature limits, ensures that both of these requirements are satisfied. Maximum fuel clad temperature limits are summarized in Table 2.2.3.

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable

design criteria specified in FSAR Chapters 2 and 3 for normal conditions. All thermal results reported in Sections 4.4.2 through 4.4.5 are within the design criteria allowable ranges for all normal conditions of storage.

#### 4.6.2 Short Term Operations~~Normal Handling and Onsite Transfer~~

*As discussed in Section 4.0, evaluation of short term operations is presented in Section 4.5. This section establishes complete compliance with the provisions of ISG-11 [4.1.4]. In particular, the ISG-11 requirement to ensure that maximum cladding temperatures under all fuel loading and short term operations be below 400 °C (752 °F) for high burnup fuel and below 570 °C (1058 °F) for moderate burnup fuel is demonstrated as stated below.*~~NUREG-1536 [4.4.10] defines several thermal acceptance criteria that are addressed in Sections 4.5.1 through 4.5.5. Each of the pertinent criteria is summarized here.~~

As required by *ISG-11*~~NUREG-1536 (4.0,IV,2)~~, the fuel cladding temperature is maintained below *the applicable limits for HBF and MBF (Table 4.3.1) during short term operations*~~570 °C (1058 °F)~~ for fuel transfer operations. Maximum clad temperatures for normal on-site transfer conditions are reported in Section 4.5.2. Maximum clad temperatures for vacuum drying conditions are reported in Section 4.5.2.1 and comply within this limit by large conservative margins.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal and off-normal conditions, assuming rupture of 1 percent and 10 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. ~~Maximum internal pressures are reported in Section 4.5.4. Design pressures are summarized in Table 2.2.1.~~

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for *all short term*~~normal (short term) fuel handling~~ operations in order to enable components to perform their intended safety functions. ~~Maximum and minimum temperatures for fuel handling operations are reported in Sections 4.5.2 and 4.5.3, respectively. Design temperature limits are summarized in Table 2.2.3.~~

~~As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.~~

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for *all short term*~~normal (short term) fuel handling~~ operations. ~~All thermal results reported in Sections 4.5.2 through 4.5.5 are within the design criteria allowable ranges for short term conditions.~~



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[4.4.12] ~~Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data~~, Holtec Report HI-992252, Rev. 1.

## APPENDIX 4.A: INTENTIONALLY DELETED

## APPENDIX 4.B: CONSERVATISMS IN THE THERMAL ANALYSIS OF THE HI-STORM 100 SYSTEM

### 4.B.1 OVERVIEW OF CASK HEAT REMOVAL SYSTEM

The HI-STORM 100 overpack is a large, cylindrical structure with an internal cavity suited for emplacement of a cylindrical canister containing spent nuclear fuel (SNF). The canister is arrayed in an upright manner inside the vertically oriented overpack. The design of the system provides for a small radial gap between the canister and the cylindrical overpack cavity. One principal function of a fuel storage system is to provide a means for ensuring fuel cladding integrity under long-term storage periods (20 years or more). The HI-STORM 100 overpack is equipped with four large ducts near its bottom and top extremities. The ducted overpack construction, together with an engineered annular space between the MPC cylinder and internal cavity in the HI-STORM 100 overpack structure, ensures a passive means of heat dissipation from the stored fuel via ventilation action (i.e., natural circulation of air in the canister-to-overpack annulus). In this manner a large structure physically interposed between the hot canister and ambient air (viz. the concrete overpack engineered for radiation protection) is rendered as an air flow device for convective heat dissipation. The pertinent design features producing the air ventilation (“chimney effect”) in the HI-STORM 100 cask are shown in Figure 4.B.1.

A great bulk of the heat emitted by the SNF is rejected to the environment ( $Q_1$ ) by convective action. A small quantity of the total heat rejection occurs by natural convection and radiation from the surface of the overpack ( $Q_2$ ), and an even smaller amount is dissipated by conduction to the concrete pad upon which the HI-STORM 100 overpack is placed ( $Q_3$ ). From the energy conservation principle, the sum of heat dissipation to all sinks (convective cooling ( $Q_1$ ), surface cooling ( $Q_2$ ) and cooling to pad ( $Q_3$ )) equals the sum of decay heat emitted from the fuel stored in the canister ( $Q_d$ ) and the heat deposited by insolation,  $Q_s$  (i.e.,  $Q_d + Q_s = Q_1 + Q_2 + Q_3$ ). This situation is illustrated in Figure 4.B.2. In the HI-STORM 100 System,  $Q_1$  is by far the dominant mode of heat removal, accounting for well over 80% of the decay heat conveyed to the external environment. Figure 4.B.3 shows the relative portions of  $Q_d$  transferred to the environs via  $Q_1$ ,  $Q_2$ , and  $Q_3$  in the HI-STORM 100 System under the design basis heat load.

The heat removal through convection,  $Q_1$ , is similar to the manner in which a fireplace chimney functions: Air is heated in the annulus between the canister and the overpack through contact with the canister’s hot cylindrical surface causing it to flow upward toward the top (exit) ducts and inducing the suction of the ambient air through the bottom ducts. The flow of air sweeping past the cylindrical surfaces of the canister has sufficient velocity to create turbulence that aids in the heat extraction process. It is readily recognized that the chimney action relies on a fundamental and immutable property of air, namely that air becomes lighter (i.e., more buoyant) as it is heated. If the canister contained no heat emitting fuel, then there would be no means for the annulus air to heat and rise. Similarly, increasing the quantity of heat produced in the canister would make more heat available for heating of annulus air, resulting in a more vigorous chimney action. Because the heat energy of the spent nuclear fuel itself actuates the chimney action,

ventilated overpacks of the HI-STORM 100 genre are considered absolutely safe against thermal malfunction. While the removal of heat through convective mass transport of air is the dominant mechanism, other minor components, labeled  $Q_2$  and  $Q_3$  in the foregoing, are recognized and quantified in the thermal analysis of the HI-STORM 100 System.

Heat dissipation from the exposed surfaces of the overpack,  $Q_2$ , occurs principally by natural convection and radiation cooling. The rate of decay heat dissipation from the external surfaces is, of course, influenced by several factors, some of which aid the process (e.g., wind, thermal turbulence of air), while others oppose it (for example, radiant heating by the sun or blocking of radiation cooling by surrounding casks). In this appendix, the relative significance of  $Q_2$  and  $Q_3$  and the method to conservatively simulate their effect in the HI-STORM 100 thermal model is discussed.

The thermal problem posed for the HI-STORM 100 System in the system's Final Safety Analysis Report (FSAR) is as follows: Given a specified maximum fuel cladding temperature,  $T_c$ , and a specified ambient temperature,  $T_a$  what is the maximum permissible heat generation rate  $Q_d$ , in the canister under steady state conditions? Of course, in the real world, the ambient temperature,  $T_a$ , varies continuously, and the cask system is rarely in a steady state (i.e., temperatures vary with time). Fortunately, fracture mechanics of spent fuel cladding instruct us that it is the time-integrated effect of elevated temperature, rather than an instantaneous peak value, that determines whether fuel cladding would rupture. The most appropriate reference ambient temperature for cladding integrity evaluation, therefore, is the average ambient temperature for the entire duration of dry storage. For conservatism, the reference ambient temperature is, however, selected to be the maximum yearly average for the ISFSI site. In the general certification of HI-STORM 100, the reference ambient temperature (formally referred to as the normal temperature) is set equal to 80°F, which is greater than the annual average for any power plant location in the U.S.\*

The thermal analysis of the cask system leads to a computed value of the fuel cladding temperature greater than  $T_a$  by an amount  $C$ . In other words,  $T_c = T_a + C$ , where  $C$  decreases slightly as  $T_a$  (assumed ambient temperature) is increased. The thermal analysis of HI-STORM 100 is carried out to compute  $C$  in a most conservative manner. In other words, the mathematical model seeks to calculate an upper bound on the value of  $C$ .

Dry storage scenarios are characterized by relatively large temperature elevations ( $C$ ) above ambient (650°F or so). The cladding temperature rise is the cumulative sum of temperature increments arising from individual elements of thermal resistance. To protect cladding from overheating, analytical assumptions adversely impacting heat transfer are chosen with particular attention given to those temperature increments which form the bulk of the temperature rise. In this appendix, the principal conservatisms in the thermal modeling of the HI-STORM 100 System and their underlying theoretical bases are presented. This overview is intended to provide

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\* According to the U.S. National Oceanic and Atmospheric Administration (NOAA) publication, "Comparative Climatic Data for the United States through 1998", the highest annual average temperature for any location in the continental U.S. is 77.8°F in Key West, Florida.

a physical understanding of the large margins buried in the HI-STORM 100 design which are summarized in Section 4.4.6 of this FSAR.

#### 4.B.2 CONSERVATISM IN ENVIRONMENTAL CONDITION SPECIFICATION

The ultimate heat sink for decay heat generated by stored fuel is ambient air. The HI-STORM 100 System defines three ambient temperatures as the environmental conditions for thermal analysis. These are, the Normal (80°F), the Off-Normal (100°F) and Extreme Hot (125°F) conditions. Two factors dictate the stipulation of an ambient temperature for cladding integrity calculations. One factor is that ambient temperatures are constantly cycling on a daily basis (night and day). Furthermore, there are seasonal variations (summer to winter). The other factor is that cladding degradation is an incremental process that, over a long period of time (20 years), has an accumulated damage resulting from an “averaged-out” effect of the environmental temperature history. The 80°F normal temperature stated in the HI-STORM 100 FSAR is defined as the highest annual average temperature at a site established from past records. This is a principal design parameter in the HI-STORM 100 analysis because it establishes the basis for demonstrating long-term SNF integrity. The choice of maximum annual average temperature is conservative for a 20-year period. Based on meteorological data, the 80°F is chosen to bound annual average temperatures reported within the continental US.

For short periods, it is recognized that ambient temperature excursions above 80°F are possible. Two scenarios are postulated and analyzed in the FSAR to bound such transient events. The Off-Normal (100°F) and Extreme Hot (125°F)\* cases are postulated as continuous (72-hour average) conditions. Both cases are analyzed as steady-state conditions (i.e., thermal inertia of the considerable concrete mass, fuel and metal completely neglected) occurring at the start of dry storage when the decay heat load to the HI-STORM 100 System is at its peak value with fuel emitting heat at its design basis maximum level.

#### 4.B.3 CONSERVATISM IN MODELING THE ISFSI ARRAY

Traditionally, in the classical treatment of the ventilated storage cask thermal problem, the cask to be analyzed (the subject cask) is modeled as a stand-alone component that rejects heat to the ambient air through chimney action ( $Q_1$ ) by natural convection to quiescent ambient air and radiation to the surrounding open spaces ( $Q_2$ ), and finally, a small amount through the concrete pad into the ground ( $Q_3$ ). The contributing effect of the sun (addition of heat) is considered, but the dissipative effect of wind is neglected. The interchange of radiative heat between proximate casks is also neglected (the so-called “cask-to-cask interactions”). In modeling the HI-STORM 100 System, Holtec International extended the classical cask thermal model to include the effect of the neighboring casks in a most conservative manner. This model represents the flow of supply air to the inlet ducts for the subject cask by erecting a cylinder around the subject cask. The model blocks all lateral flow of air from the surrounding space into the subject cask’s inlet ducts. This mathematical artifice is illustrated in Figure 4.B.4, where the lateral air flow arrows are shown “dotted” to indicate that the mathematical cylinder constructed around the cask has

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\* According to NOAA, the highest daily mean temperature for any location in the continental U.S. is 93.7°F, which occurred in Yuma, Arizona.

blocked off the lateral flow of air. Consequently, the chimney air must flow down the annulus from the air plenum space above the casks, turn around at the bottom and enter the inlet ducts. Because the vertical downflow of air introduces additional resistance to flow, an obvious effect of the hypothetical enclosing cylinder construct is an increased total resistance to the chimney flow which, it is recalled, is the main heat conveyance mechanism in a ventilated cask. Throttling of the chimney flow by the hypothetical enclosing cylinder is an element of conservatism in the HI-STORM modeling.

Thus, whereas air flows toward the bottom ducts from areas of supply which are scattered in a three dimensional continuum with partial restriction from neighboring casks, the analytical model blocks the air flow completely from areas outside the hypothetical cylinder. This is illustrated in Figure 4.B.4 in which an impervious boundary is shown to limit HI-STORM 100 cask access to fresh air from an annular opening near the top.

Thus, in the HI-STORM model, the feeder air to the HI-STORM 100 System must flow down the hypothetical annulus sweeping past the external surface of the cask. The ambient air, assumed to enter this hypothetical annulus at the assumed environmental temperature, heats by convective heat extraction from the overpack before reaching the bottom (inlet) ducts. In this manner, the temperature of the feeder air into the ducts is maximized. In reality, the horizontal flow of air in the vicinity of the inlet ducts, suppressed by the enclosed cylinder construct (as shown in Figure 4.B.4) would act to mitigate the pre-heating of the feeder air. By maximizing the extent of air preheating, the computed value of ventilation flow is underestimated in the simulation.

#### 4.B.4 CONSERVATISM IN RADIANT HEAT LOSS

In an array of casks, the external (exposed) cask surfaces have a certain “view” of each other. The extent of view is a function of relative geometrical orientation of the surfaces and presence of other objects between them. The extent of view influences the rate of heat exchange between surfaces by thermal radiation. The presence of neighboring casks also partially blocks the escape of radiant heat from a cask thus affecting its ability to dissipate heat to the environment. This aspect of Radiative Blocking (RB) is illustrated for a reference cask (shown shaded) in Figure 4.B.5. It is also apparent that a cask is a recipient of radiant energy from adjacent casks (Radiant Heating (RH)). Thus, a thermal model representative of a cask array must address the RB and RH effects in a conservative manner. To bound the physical situation, a Hypothetical Reflecting Boundary (HRB) modeling feature is introduced in the thermal model. The HRB feature surrounds the HI-STORM 100 overpack with a reflecting cylindrical surface with the boundaries insulated.

In Figures 4.B.6 and 4.B.7 the inclusion of RB and RH effects in the HI-STORM 100 modeling is graphically illustrated. Figure 4.B.6 shows that an incident ray of radiant energy leaving the cask surface bounces back from the HRB thus preventing escape (i.e., RB effect maximized). The RH effect is illustrated in Figure 4.B.7 by superimposing on the physical model reflected images of HI-STORM 100 cask surrounding the reference cask. A ray of radiant energy from an adjacent cask directed toward the reference cask (AA) is duplicated by the model via another ray



of radiant energy leaving the cask (BB) and being reflected back by the HRB (BA'). A significant feature of this model is that the reflected ray (BA') initiated from a cask surface (reference cask) assumed to be loaded with design basis maximum heat (hottest surface temperature). As the strength of the ray is directly proportional to the fourth power of surface temperature, radiant energy emission from an adjacent cask at a lower heat load will be overestimated by the HRB construct. In other words, the reference cask is assumed to be in an array of casks all producing design basis maximum heat. Clearly, it is physically impossible to load every location of every cask with fuel emitting heat at design basis maximum. Such a spent fuel inventory does not exist. This bounding assumption has the effect of maximizing cask surface temperature as the possibility of "hot" (design basis) casks being radiatively cooled by adjacent casks is precluded. The HRB feature included in the HI-STORM 100 model thus provides a bounding effect of an infinite array of casks, all at design basis maximum heat loads. No radiant heat is permitted to escape the reference cask (bounding effect) and the reflecting boundary mimics incident radiation toward the reference casks around the 360° circumference (bounding effect).

#### 4.B.5 CONSERVATISM IN REPRESENTING BASKET AXIAL RESISTANCE

As stated earlier, the largest fraction of the total resistance to the flow of heat from the spent nuclear fuel (SNF) to the ambient is centered in the basket itself. Out of the total temperature drop of approximately 650°F (C=650°F) between the peak fuel cladding temperature and the ambient, over 400°F occurs in the fuel basket. Therefore, it stands to reason that conservatism in the basket thermal simulation would have a pronounced effect on the conservatism in the final solution. The thermal model of the fuel basket in the HI-STORM 100 FSAR was accordingly constructed with a number of conservative assumptions that are described in the HI-STORM 100 FSAR. We illustrate the significance of the whole array of conservatisms by explaining one in some detail in the following discussion.

It is recognized that the heat emission from a fuel assembly is axially non-uniform. The maximum heat generation occurs at about the mid-height region of the enriched uranium column, and tapers off toward its extremities. The axial heat conduction in the fuel basket would act to diffuse and levelize the temperature field in the basket. The axial conductivity of the basket, quite clearly, is the key determinant in how well the thermal field in the basket would be homogenized. It is also evident that the conduction of heat along the length of the basket occurs in an uninterrupted manner in a HI-STORM 100 basket because of its continuously welded honeycomb geometry. On the other hand, the in-plane transfer of heat must occur through the physical gaps that exist between the fuel rods, between the fuel assembly and the basket walls and between the basket and the MPC shell. These gaps depress the in-plane conductivity of the basket. However, in the interest of conservatism, only a small fraction of the axial conductivity of the basket is included in the HI-STORM 100 thermal model. This assumption has the direct effect of throttling the axial flow of heat and thus of elevating the computed value of mid-height cladding temperature (where the peak temperature occurs) above its actual value. In actuality, the axial conductivity of the fuel basket is much greater than the in-plane conductivity due to the continuity of the fuel and basket structures in that direction. Had the axial conductivity of the basket been modeled less conservatively in the HI-STORM 100 thermal analysis, then the

temperature distribution in the basket will be more uniform, i.e., the bottom region of the basket would be hotter than that computed. This means that the temperature of the MPC's external surface in the bottom region is hotter than computed in the HI-STORM 100 analysis. It is a well-known fact in ventilated column design that the lower the location in the column where the heat is introduced, the more vigorous the ventilation action. Therefore, the conservatism in the basket's axial conductivity assumption has the net effect of reducing the computed ventilation rate.

To estimate the conservatism in restricting the basket axial resistance, we perform a numerical exercise using mathematical perturbation techniques. The axial conductivity ( $K_z$ ) of the MPC is, as explained previously, much higher than the in-plane ( $K_r$ ) conductivity. The thermal solution to the MPC anisotropic conductivities problem (i.e.  $K_z$  and  $K_r$  are not equal) is mathematically expressed as a sum of a baseline isotropic solution  $T_o$  (setting  $K_z = K_r$ ) and a perturbation  $T^*$  which accounts for anisotropic effects. From Fourier's Law of heat conduction in solids, the perturbation equation for  $T^*$  is reduced to the following form:

$$K_z \frac{d^2 T^*}{dz^2} = -\Delta K \frac{d^2 T_o}{dz^2}$$

Where,  $\Delta K$  is the perturbation parameter (i.e. axial conductivity offset  $\Delta K = K_z - K_r$ ). The boundary conditions for the perturbation solution are zero slope at peak cladding temperature location ( $dT^*/dz = 0$ ) (which occurs at about the top of the active fuel height) and  $T^* = 0$  at the bottom of the active fuel length. The object of this calculation is to compute  $T^*$  where the peak fuel cladding temperature is reached. To this end, the baseline thermal solution  $T_o$  (i.e. HI-STORM isotropic modeling solution) is employed to compute an appropriate value for  $d^2 T_o/dz^2$  which characterizes the axial temperature rise over the height of the active fuel length in the hottest fuel cell. This is computed as  $(-\Delta T_{ax}/L^2)$  where  $\Delta T_{ax}$  is the fuel cell temperature rise and  $L$  is the active fuel length. Conservatively postulating a lower bound  $\Delta T_{ax}$  of 200°F and  $L$  of 12 ft,  $d^2 T_o/dz^2$  is computed as  $-1.39^\circ\text{F}/\text{ft}^2$ . Integrating the perturbation equation shown above, the following formula for  $T^*$  is obtained:

$$T^* = \left( \frac{\Delta K}{K_z} \right) \frac{d^2 T_o}{dz^2} L^2$$

Employing a conservative low value for the  $(\Delta K/K_z)$  parameter of 0.15,  $T^*$  is computed as  $-30^\circ\text{F}$ . In other words, the baseline HI-STORM solution over predicts the peak cladding temperature by approximately  $30^\circ\text{F}$ .

#### 4.B.6 HEAT DISSIPATION UNDERPREDICTION IN THE MPC DOWNCOMER

Internal circulation of helium in the sealed MPC is modeled as flow in a porous medium in the fueled region containing the SNF (including top and bottom plenums). The basket-to-MPC shell clearance space is modeled as a helium filled radial gap to include the downcomer flow in the thermal model. The downcomer region, as illustrated in Figure 4.4.2, consists of an azimuthally varying gap formed by the square-celled basket outline and the cylindrical MPC shell. At the locations of closest approach a differential expansion gap (a small clearance on the order of 1/10 of an inch) is engineered to allow free thermal expansion of the basket. At the widest locations, the gaps are on the order of the fuel cell opening ( $\sim 6''$  (BWR) and  $\sim 9''$  (PWR) MPCs). It is

heuristically evident that heat dissipation by conduction is maximum at the closest approach locations (low thermal resistance path) and that convective heat transfer is highest at the widest gap locations (large downcomer flow). In the FLUENT thermal model, a radial gap that is large compared to the basket-to-shell clearance and small compared to the cell opening is used. As a relatively large gap penalizes heat dissipation by conduction and a small gap throttles convective flow, the use of a single gap in the FLUENT model understates both conduction and convection heat transfer in the downcomer region. ~~Furthermore, heat dissipation by the aluminum heat conduction elements, if used, is conservatively neglected in the thermosiphon models employed in the HI-STORM modeling.~~

Heat dissipation in the downcomer region is the sum of ~~five~~*four* elements, viz. convective heat transfer (C1), helium conduction heat transfer (C2), basket-to-shell contact heat transfer (C3), *and* radiation heat transfer (C4) ~~and aluminum conduction elements (if used) heat transfer (C5).~~ In the HI-STORM thermal modeling, ~~two~~*one* elements of heat transfer (C3 ~~and~~ C5) *is* completely neglected, C2 is severely penalized and C1 is underpredicted. In other words the HI-STORM thermosiphon model has choked the radial flow of heat in the downcomer space. This has the direct effect of raising the temperature of fuel in the thermal solutions.

#### 4.B.7 CONSERVATISM IN MPC EXTERNAL HEAT DISSIPATION TO CHIMNEY AIR

The principle means of decay heat dissipation to the environment is by cooling of the MPC surface by chimney air flow. Heat rejection from the MPC surface is by a combination of convective heat transfer to a through flowing fluid medium (air), natural convection cooling at the outer overpack surface, and by radiation heat transfer. Because the temperature of the fuel stored in the MPC is directly affected by the rate of heat dissipation from the canister external surface, heat transfer correlations with robust conservatisms are employed in the HI-STORM simulations. The FLUENT computer code deployed for the modeling employs a so called “wall-functions” approach for computing the transfer of heat from solid surfaces to fluid medium. This approach has the desired effect of computing heat dissipation in a most conservative manner. As this default approach has been employed in the thermal modeling, it is contextually relevant to quantify the conservatism in a classical setting to provide an additional level of assurance in the HI-STORM results. To do this, we have posed a classical heat transfer problem of a heated square block cooled in a stream of upward moving air. The problem is illustrated in Figure 4.B.8. From the physics of the problem, the maximum steady state solid interior temperature ( $T_{\max}$ ) is computed as:

$$T_{\max} = T_{\text{sink}} + \Delta T_{\text{air}} + \Delta T_{\text{s}}$$

where,

$T_{\text{sink}}$	= Sink temperature (mean of inlet and outlet air temperature)
$\Delta T_{\text{air}}$	= Solid surface to air temperature difference
$\Delta T_{\text{s}}$	= Solid block interior temperature elevation

The sink temperature is computed by first calculating the air outlet temperature from energy conservation principles. Solid-to-air heat transfer is computed using classical natural convection correlation proposed by Jakob and Hawkins (“Elements of Heat Transfer”, John Wiley & Sons,

1957) and  $\Delta T$ s is readily computed by an analytical solution to the equation of heat conduction in solids. By solving this same problem on the FLUENT computer code using the in-built “wall-functions”, in excess of 100°F conservative margin over the classical result for  $T_{\max}$  is established.

#### 4.B.8 MISCELLANEOUS CONSERVATISMS

Section 4.4.6 of the FSAR lists eleven elements of conservatism, of which certain non-transparent and individually significant items are discussed in detail in this appendix. Out of the balance of conservatisms, the one of notable mention is the conservatism in fuel decay heat generation stipulation based on the most heat emissive fuel assembly type. This posture imputes a large conservatism for certain other fuel types, which have a much lower quantity of Uranium fuel inventory relative to the design basis fuel type. Combining this with other miscellaneous conservatisms, an aggregate effect is to overestimate cladding temperatures by about 15°F to 50°F.

#### 4.B.9 CONCLUSIONS

The foregoing narrative provides a physical description of the many elements of conservatism in the HI-STORM 100 thermal model. The conservatisms may be broadly divided into two categories:

1. Those intrinsic to the FLUENT modeling process.
2. Those arising from the input data and on the HI-STORM 100 thermal modeling.

The conservatism in Category (1) may be identified by reviewing the Holtec International Benchmark Report [4.B.1], which shows that the FLUENT solution methodology, when applied to the prototype cask (TN 24P) over-predicts the peak cladding temperature by as much as 79 °F. and as much as 37°F relative to the PNNL results (see Attachment 1 to Reference [4.B.1]) from their COBRA SFS solution as compared against Holtec’s FLUENT solution.

Category (2) conservatisms are those that we have deliberately embedded in the HI-STORM 100 thermal model to ensure that the computed value of the peak fuel cladding temperature is further over-stated. Table 4.B.1 contains a listing of the major conservatisms in the HI-STORM 100 thermal model, along with an estimate of the effect (increase) of each on the computed peak cladding temperature.

**Table 4.B.1**

**Conservatism in the HI-STORM 100 Thermal Model**

MODELING ELEMENT	CONSERVATISM [°F]
Long Term Ambient Temperature	2 to 30
Hypothetical Cylinder Construct	~5
Axial Heat Dissipation Restriction	30
MPC Downcomer Heat Dissipation Restriction	50
MPC External Heat Dissipation Under-prediction	50
Miscellaneous Conservatisms	15 to 50

4.B.9 REFERENCES

- [4.B.1] “Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data”, Holtec Report HI-992252, Rev. 1.

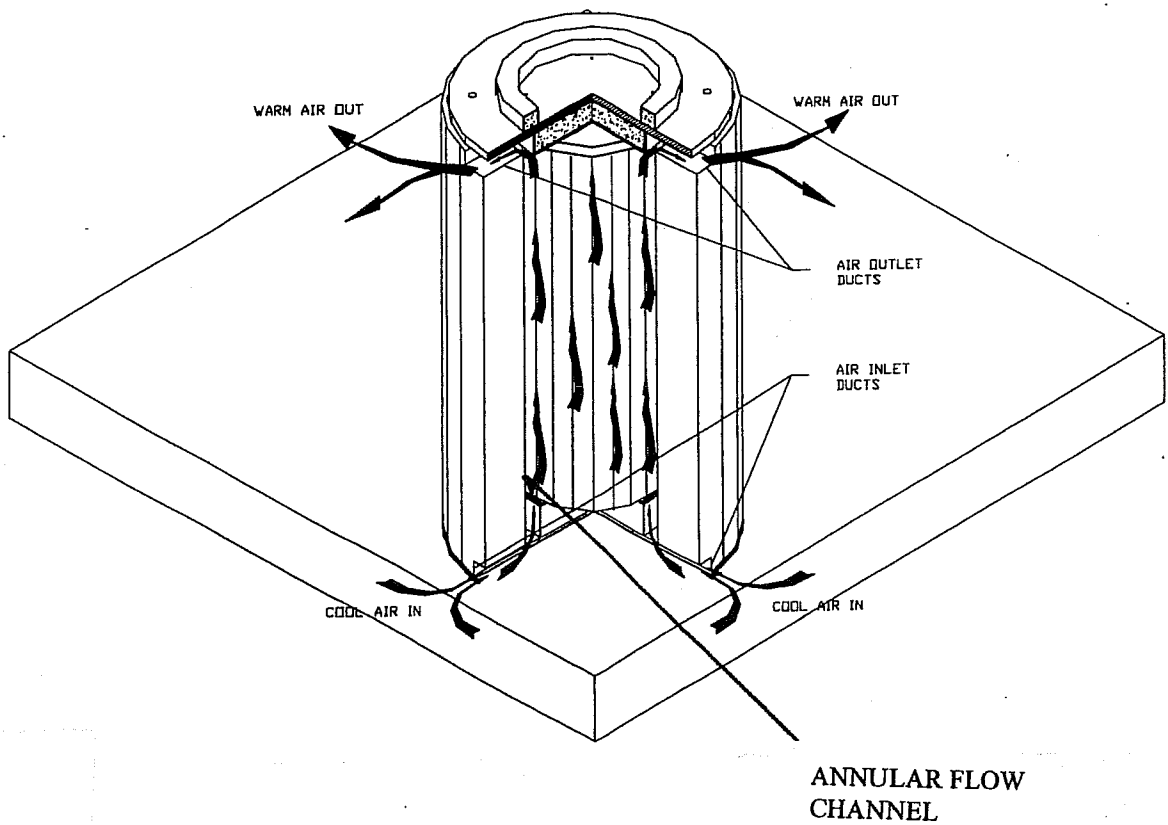


FIGURE 4.B.1: CUTAWAY VIEW OF A HI-STORM OVERPACK  
STANDING ON AN ISFSI PAD

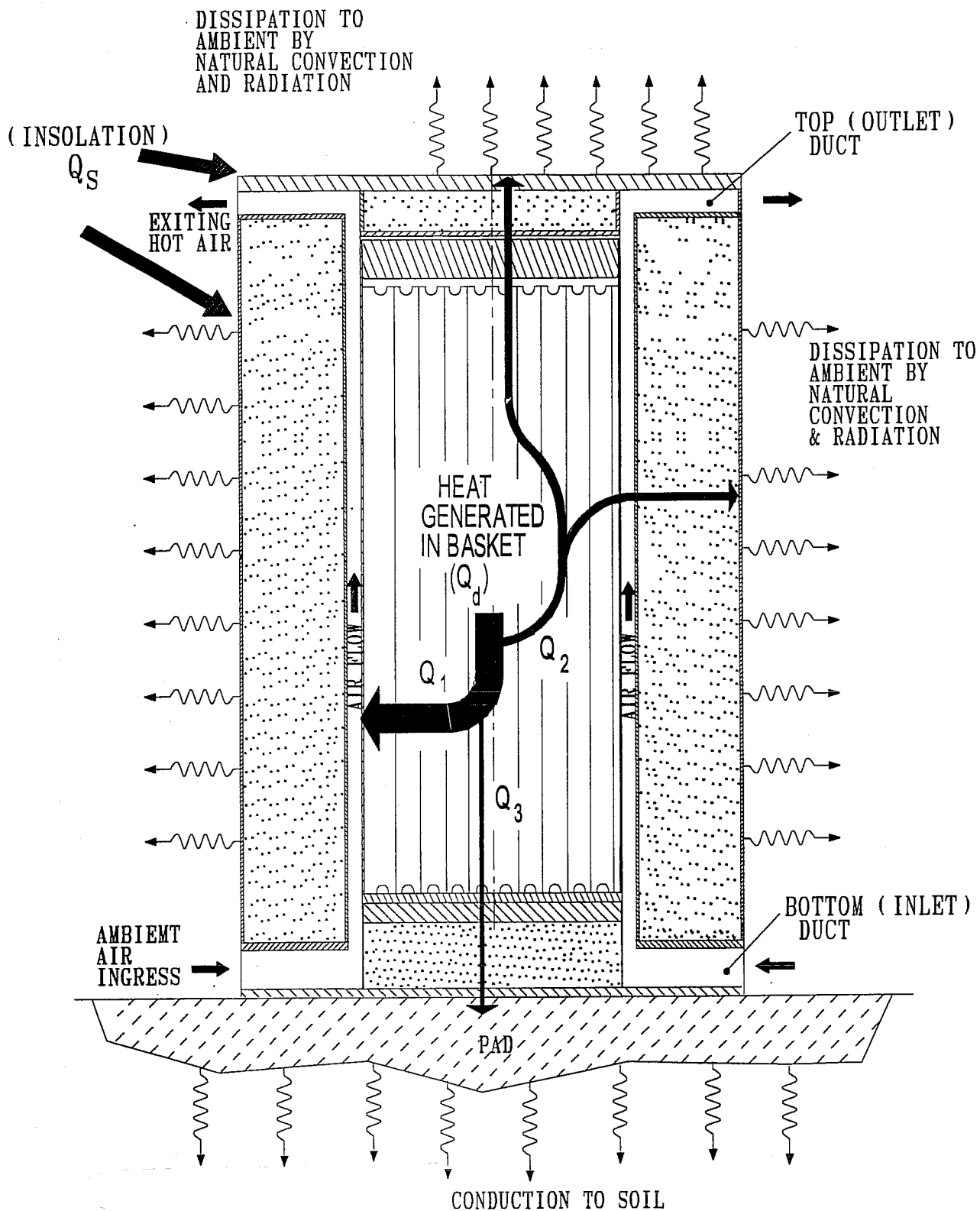


FIGURE 4.B.2: DEPICTION OF THE HI-STORM VENTILATED CASK HEAT DISSIPATION ELEMENTS

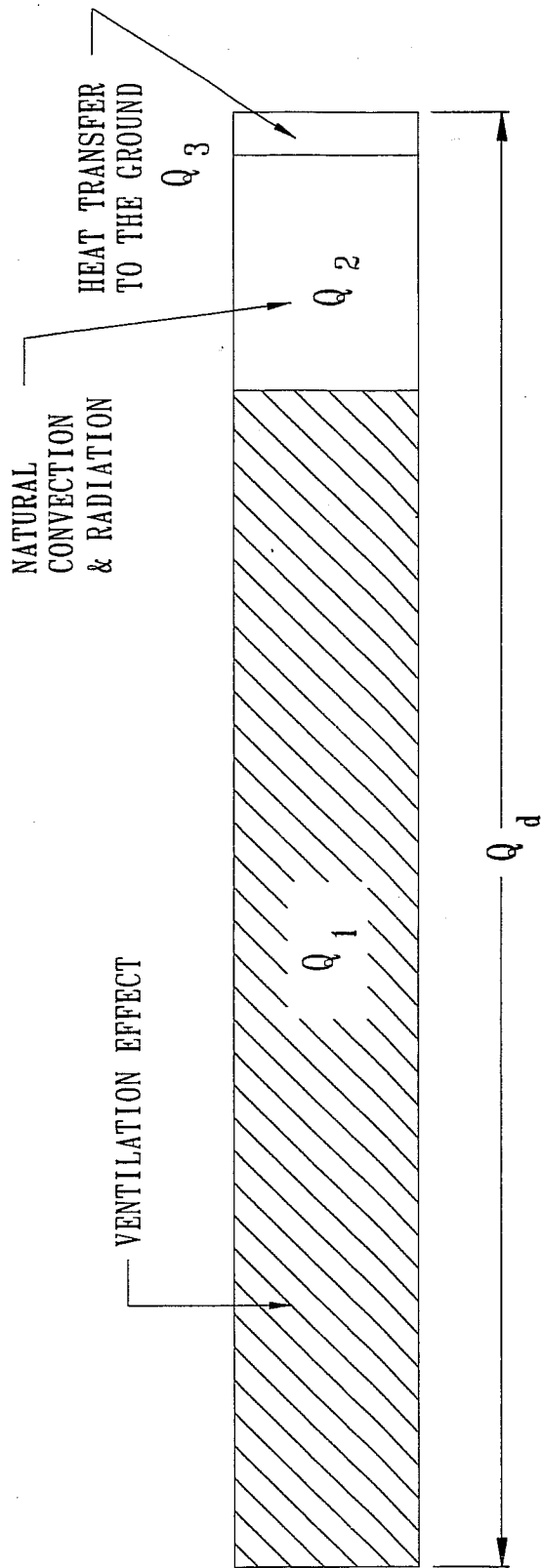



FIGURE 4.B.3: RELATIVE SIGNIFICANCE OF HEAT DISSIPATION ELEMENTS IN THE HI-STORM 100



LEGEND:  IMPERVIOUS BOUNDARY

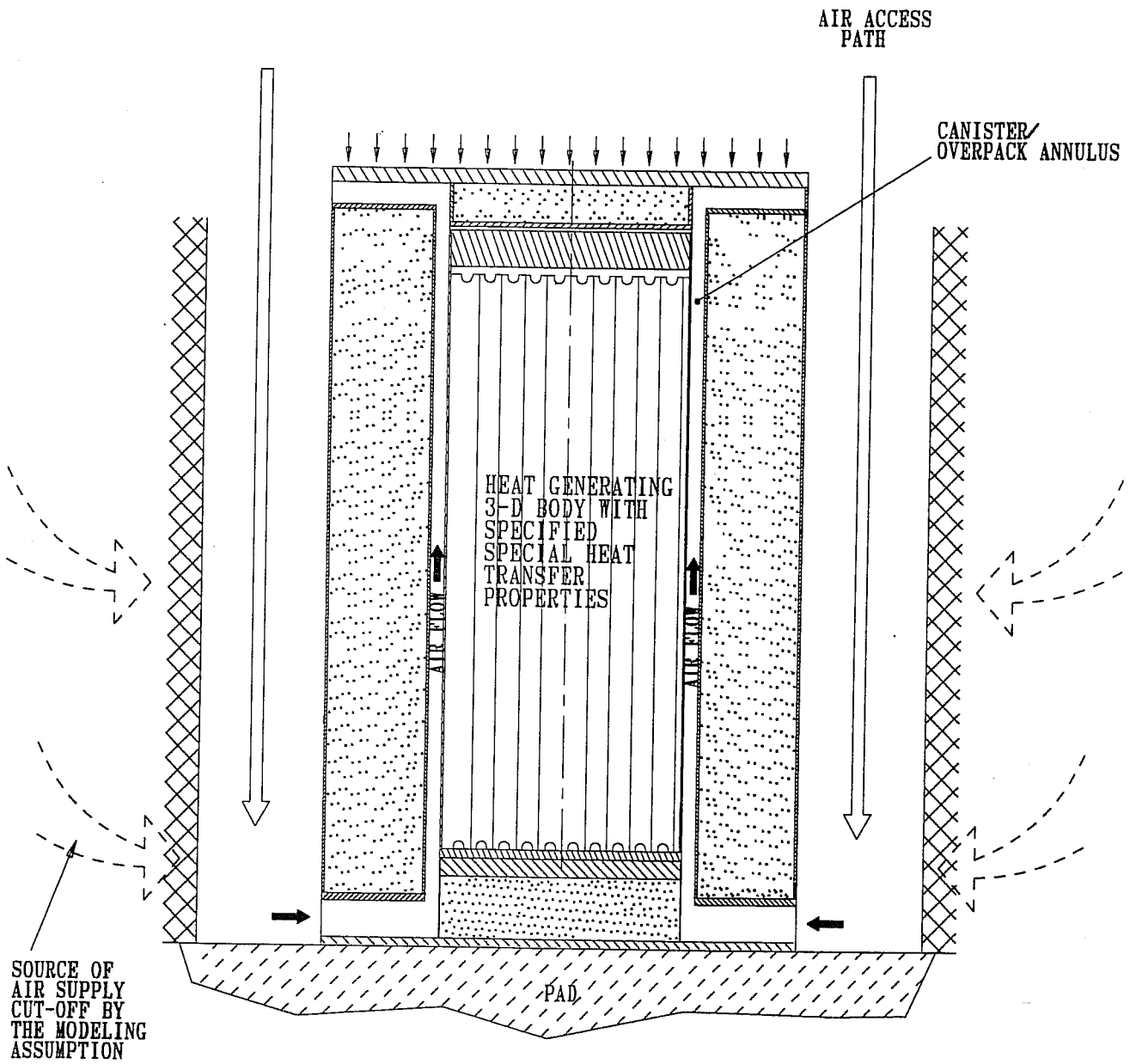


FIGURE 4.B.4: AIR ACCESS RESTRICTIONS IN THE HI-STORM THERMAL MODEL

REVISION 1

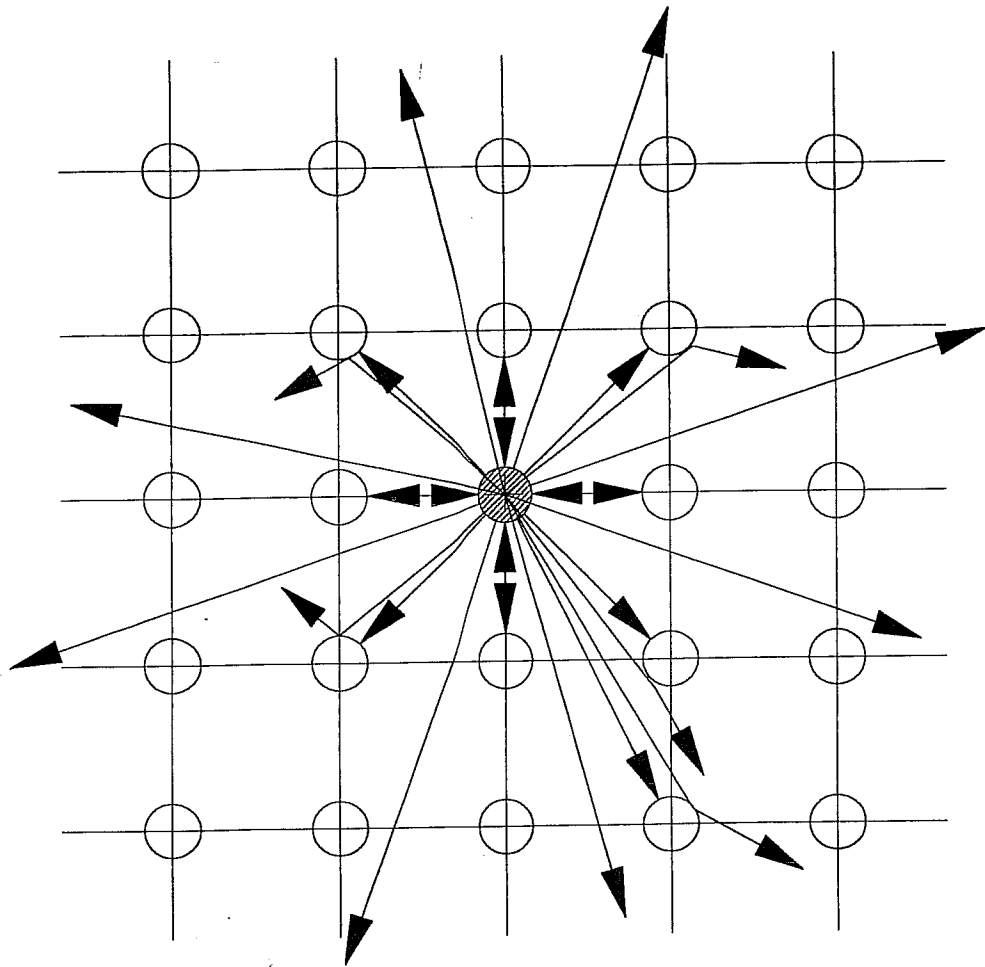


FIGURE 4.B.5: IN-PLANE RADIATIVE COOLING OF A HI-STORM CASK IN AN ARRAY

REVISION 1

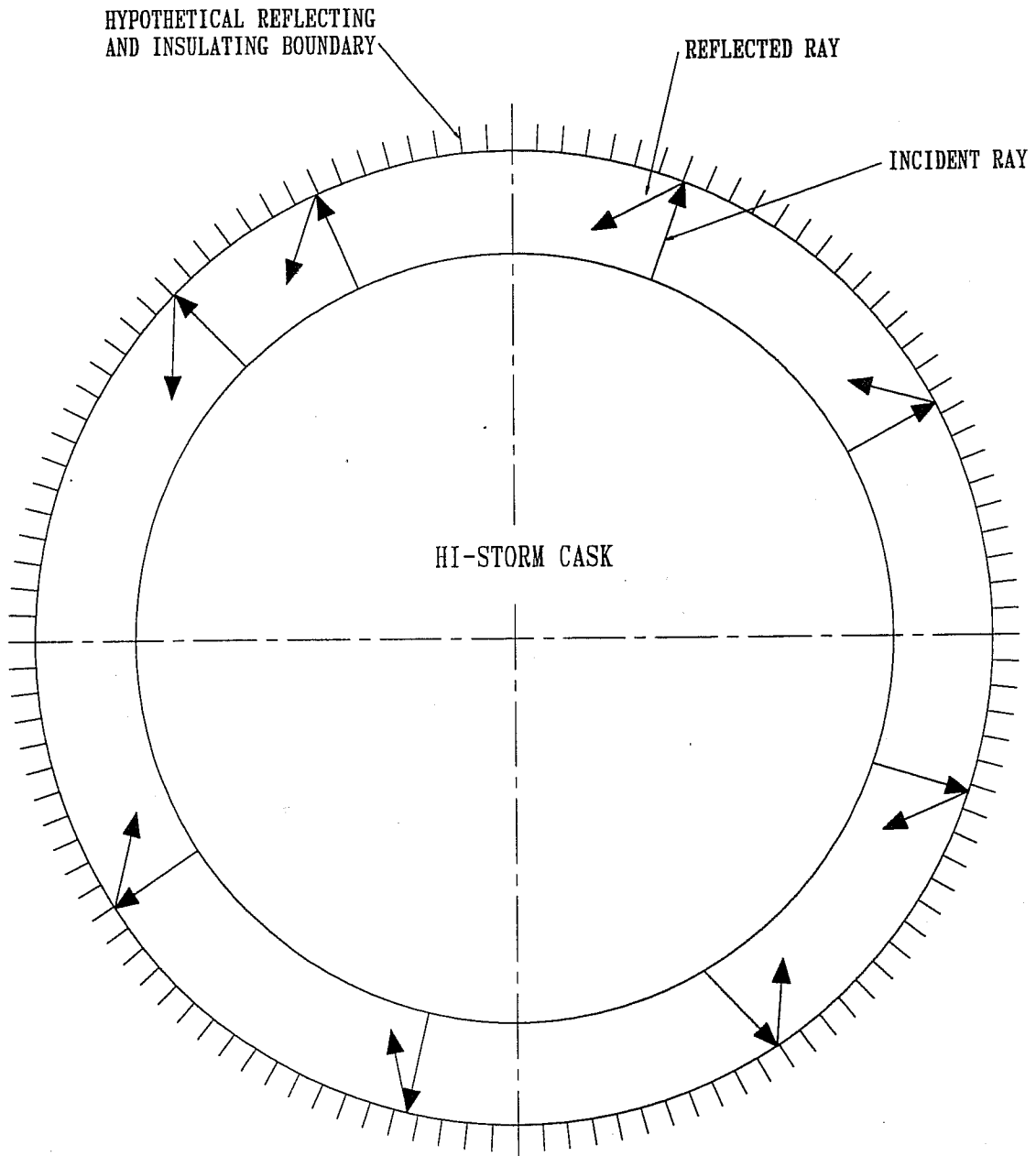


FIGURE 4.B.6: IN-PLANE RADIATIVE BLOCKING OF A HI-STORM CASK BY HYPOTHETICAL REFLECTING BOUNDARY

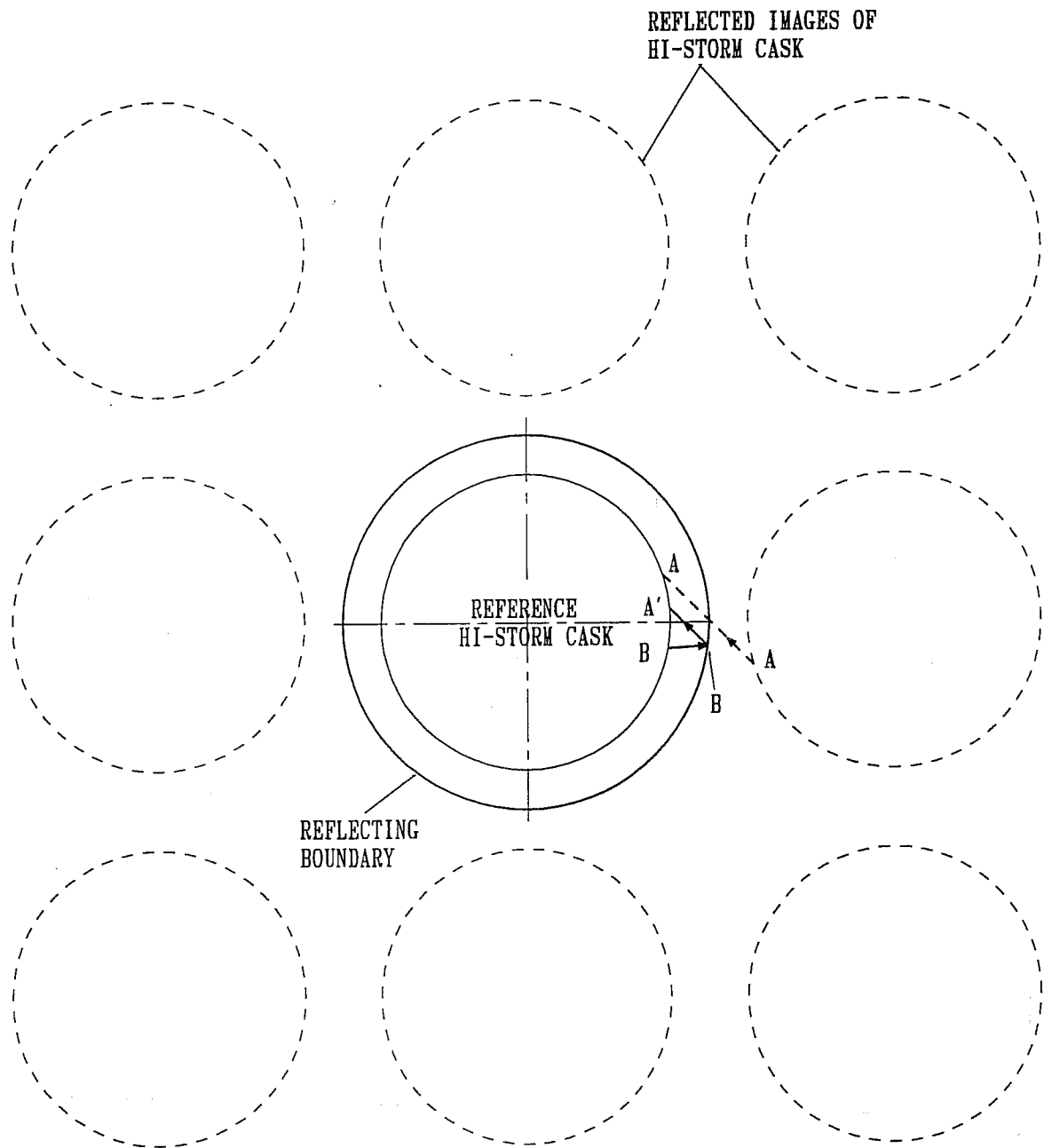


FIGURE 4.B.7: RADIATIVE HEATING OF REFERENCE HI-STORM CASK BY SURROUNDING CASKS

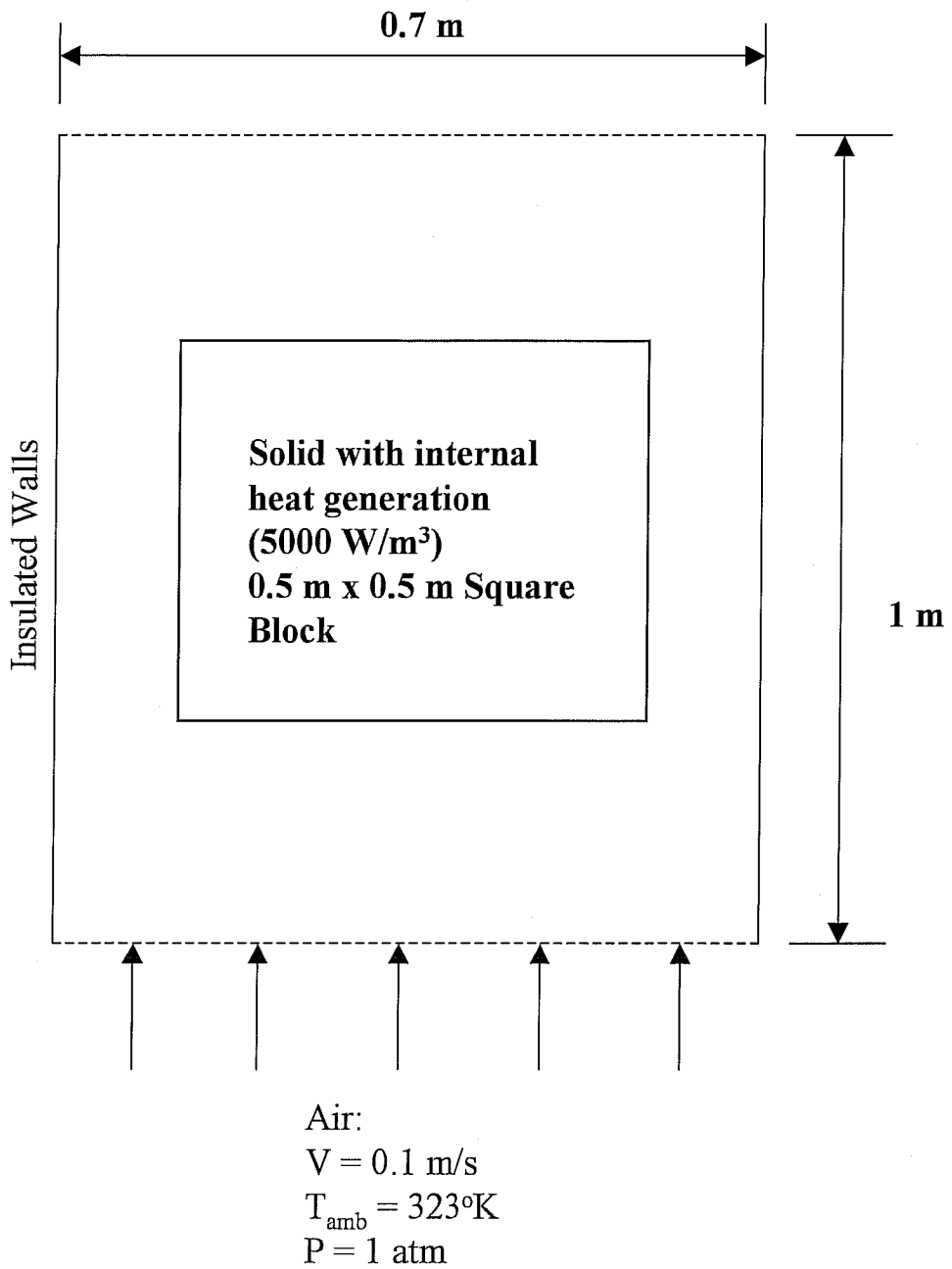


FIGURE 4.B.8: A CLASSICAL THERMAL SCENARIO: AIR COOLING OF A HEATED SQUARE BLOCK

## CHAPTER 5<sup>†</sup>: SHIELDING EVALUATION

### 5.0 INTRODUCTION

The shielding analysis of the HI-STORM 100 System, including the HI-STORM 100 overpack, HI-STORM 100S overpack, and the 100-ton 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 two HI-STORM overpacks (the HI-STORM 100S overpack is a shorter version of the HI-STORM 100 overpack). 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 ~~the approved contents section of Appendix B to the CoC 2.1.9.~~ Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. DFCs containing BWR fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing BWR damaged fuel assemblies may be stored in either the MPC-68, the MPC-68F, or the MPC-68FF. DFCs containing PWR fuel debris must be stored in the MPC-24EF or MPC-32F while DFCs containing PWR damaged fuel assemblies may be stored in either the MPC-24E, ~~or MPC-24EF,~~ MPC-32, or MPC-32F.

The MPC-68, MPC-68F, and MPC-68FF 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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<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter I*, 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).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs) or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices.

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 System offers two different loading patterns in the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F*, MPC-68, and the MPC-68FF. These patterns are uniform and regionalized loading as described in Section 2.0.1 and 2.1.6. Since the different loading patterns have different allowable burnup and cooling times combinations, both loading patterns are discussed in this chapter.

The sections that follow will demonstrate that the design of the HI-STORM 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

#### Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The “controlled area” is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.
2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.
3. Dose rates from the cask must be consistent with a well established “as low as reasonably achievable” (ALARA) program for activities in and around the storage site.
4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR 72.106.
5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- The HI-STORM 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STORM 100 System is acceptable during normal, off-normal, and accident conditions.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains *a discussion on the release of radioactive materials from the HI-STORM 100 System.* ~~an analysis of the estimated dose at the controlled area boundary during normal, off-normal, and accident conditions from the release of radioactive materials.~~ Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM 100 System.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.



## 5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM 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.  $\alpha$ ,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4.  $\gamma$ ,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  $^{10}\text{B}$  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. ~~Table 2.1.6~~ *Section 2.1.9* specifies the acceptable intact zircaloy clad fuel characteristics for storage. ~~Table 2.1.7~~ *specifies and* the acceptable damaged fuel characteristics ~~for storage.~~

The design basis stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. ~~Table 2.1.8~~ *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. ~~The approved contents section of Appendix B to the CoC~~Section 2.1.9 specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in these MPCs. ~~Appendix B to the CoC~~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 ~~Appendix B to the CoC~~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 dose rates surrounding the HI-STORM overpack are very low, and thus, the shielding analysis of the HI-STORM overpack conservatively considered the burnup and cooling time combinations listed below, which bound the acceptable burnup levels and cooling times from ~~Appendix B to the CoC~~Section 2.1.9. This large conservatism is included in the analysis of the HI-STORM overpack to unequivocally demonstrate that the HI-STORM overpack meets the Part 72 dose requirements.

Zircaloy Clad Fuel		
MPC-24	MPC-32	MPC-68
5247,500 MWD/MTU 5-3 year cooling	4535,000 MWD/MTU 5-3 year cooling	47,500-40,000 MWD/MTU 5-3 year cooling
Stainless Steel Clad Fuel		
MPC-24	MPC-32	MPC-68
40,000 MWD/MTU 8 year cooling	40,000 MWD/MTU 9 year cooling	22,500 MWD/MTU 10 year cooling

The burnup and cooling time combinations analyzed for zircaloy clad fuel produce dose rates at the midplane of the HI-STORM overpack which bound all uniform and regionalized loading burnup and cooling time combinations listed in ~~Appendix B to the CoC~~ *Section 2.1.9*. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

The dose rates surrounding the HI-TRAC transfer cask are significantly higher than the dose rates surrounding the HI-STORM overpack, and although no specific regulatory limits are defined, dose rates are based on the ALARA principle. Therefore, the cited dose rates were based on ~~the actual burnups and cooling times requested in Appendix B to the CoC~~ *closer to the combinations in Section 2.1.9*. Two different burnup and cooling times, listed below, were analyzed for the MPC-24, MPC-32, and the MPC-68 in the 100-ton HI-TRAC. *The burnup and cooling time combinations were chosen for the minimum cooling time and a bounding burnup corresponding to the 14x14A in the MPC-24 and MPC-32 and the 9x9G fuel assembly in the MPC-68.* The burnups corresponding to 53-year cooling times produce dose rates at 1 meter from the radial surface of the overpack, for the locations reported in this chapter, which bound the dose rates from all other uniform loading burnup and cooling time combinations ~~listed in Appendix B to the CoC~~ *Section 2.1.9*. ~~Since it is reasonable to assume that the majority of fuel which will be loaded in casks will be 10 years or older, the dose rates from conservative burnups for 10-year cooling are also presented in this chapter.~~

100-ton HI-TRAC		
MPC-24	MPC-32	MPC-68
<del>42,500</del> 46,000 MWD/MTU 53 year cooling	<del>32,500</del> 35,000 MWD/MTU 53 year cooling	4039,000 MWD/MTU 53 year cooling
<del>52,500</del> 75,000 MWD/MTU 40-5 year cooling	<del>45,000</del> 75,000 MWD/MTU 40-8 year cooling	5070,000 MWD/MTU 40-6 year cooling

The 100-ton HI-TRAC with the MPC-24 has higher dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results for 53-year cooling are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The MPC-32 results, MPC-68 results, and additional MPC-24 results are provided in Section 5.4 for comparison.

The 100-ton HI-TRAC dose rates bound the HI-TRAC 125 and 125D dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the HI-TRAC 125 and 125D. Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. Therefore, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter. Dose rates are presented for two different burnup and cooling time combinations for the MPC-24 in the HI-TRAC 125D *which bound the allowable contents in Section 2.1.9*: ~~42,500~~46,000 MWD/MTU with 53-year cooling and ~~57,500~~75,000

MWD/MTU with 125-year cooling. The dose rates for the later combination are presented in this section because it produces the highest dose rate at the cask midplane. Dose rates for the other burnup and cooling time combination are presented in Section 5.4.

As a general statement, the dose rates for uniform loading presented in this chapter bound the dose rates for regionalized loading at 1 meter distance from the overpack. Therefore, dose rates for specific burnup and cooling time combinations in a regionalized loading pattern are not presented in this chapter. Section 5.4.9 provides an additional brief discussion on regionalized loading.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

#### 5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.
2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM 100 System in Section 2.3.5.2 as: ~~60–100~~ mrem/hour on the radial surface of the overpack, 60 mrem/hour at the openings of the air vents, and 60 mrem/hour on the top of the overpack.

The HI-STORM overpack dose rates presented in this section are conservatively evaluated for the MPC-32, the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in ~~Appendix B to the CoC~~ Section 2.1.9.

Figure 5.1.1 and 5.1.12 identify the locations of the dose points referenced in the dose rate summary tables for the HI-STORM 100 and HI-STORM 100S overpacks, respectively. Dose

Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. Dose Point #4a in Figure 5.1.12 is located directly above the exit duct and next to the concrete shield block. The dose values reported at the locations shown on Figure 5.1.1 and 5.1.12 are averaged over a region that is approximately 1 foot in width.

The total dose rates presented in this chapter for the MPC-24 and MPC-32 are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Section 5.2.4.1. This is conservative because it is not expected that the cooling times for both the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values.

Tables 5.1.1 and 5.1.3 provide the maximum dose rates adjacent to the HI-STORM 100S overpack during normal conditions for the MPC-32 and MPC-68. Tables 5.1.4 and 5.1.6 provide the maximum dose rates at one meter from the HI-STORM 100S overpack. Tables 5.1.2 and 5.1.5 provide the maximum dose rates adjacent to and one meter from the HI-STORM 100 overpack for the MPC-24.

Although the dose rates for the MPC-32 in HI-STORM 100s are ~~equivalent to or~~ greater than those for the MPC-24 in HI-STORM 100 *at the ventilation ducts*, as shown in Tables 5.1.1, 5.1.2, 5.1.4, and 5.1.5, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary. *This is acceptable because the vents are a small fraction of the radial surface area. As such, the dominant effect on the dose at distance is the radial portion of the overpack between the vents which comprises approximately 91% of the total radial surface area compared to approximately 1.3% for the vents.* The MPC-24 was chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the MPC-32 or the MPC-68 (see ~~Appendix B to the CoC~~ Section 2.1.9). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32. The dose rates at the controlled area boundary were calculated for the HI-STORM 100 overpack rather than the HI-STORM 100S overpack. The difference in height will have little impact on the dose rates at the controlled area boundary since the surface dose rates are very similar. The controlled area boundary dose rates were also calculated ~~without~~ including *the BPA non-fuel hardware source. In the site specific dose analysis, users should perform an analysis which properly bounds the fuel to be stored including BPRAs if present. This is acceptable because the dose rates for the HI-STORM 100 overpack calculated in Table 5.1.2 without BPRAs are conservative enough to bound the dose rates for actual burnup and cooling times from Appendix B to the CoC including BPRAs.*

Table 5.1.7 provides dose rates adjacent to and one meter from the 100-ton HI-TRAC. Table 5.1.8 provides dose rates adjacent to and one meter from the 125-ton HI-TRACs. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125 and 100 transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8

correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point #5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed burnup and cooling times specified in ~~Appendix B to the CoC~~Section 2.1.9, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen ~~based on~~ *bound* the allowable burnup and cooling times in ~~Appendix B to the CoC~~Section 2.1.9. Results for other burnup and cooling times and for the MPC-68 and MPC-32 are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRACs or the HI-STORM overpack, it is important to understand the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. The figures discussed below were all calculated without the gamma source from BPRAs and were calculated for an earlier design of the HI-TRAC which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below.

Figure 5.1.5 shows the dose rate profile at 1 foot from the side of the 100-ton HI-TRAC transfer cask with the MPC-24 for 35,000 MWD/MTU and 5 year cooling. This figure clearly shows the behavior of the total dose rate and each of the dose components as a function of the cask height. To capture the effect of scattering off the concrete floor, the calculational model simulates the 100-ton HI-TRAC at a height of 6 inches (the typical cask carry height) above the concrete floor. As expected, the total dose rate on the side near the top and bottom is dominated by the Co-60 gamma dose component, while the center dose rate is dominated by the fuel gamma dose component.

The total dose rate and individual dose rate components on the surface of the pool lid on the 100-ton HI-TRAC are provided in Figure 5.1.6, illustrating the significant reduction in dose rate with increasing distance from the center of the pool lid. Specifically, the total dose rate is shown to drop by a factor of more than 20 from the center of the pool lid to the outer edge of the HI-TRAC. Therefore, even though the dose rate in Table 5.1.7 at the center of the pool lid is substantial, the dose rate contribution, from the pool lid, to the personnel exposure is minimal.

The behavior of the dose rate 1-foot from the transfer lid is shown in Figure 5.1.7. Similarly, the total dose rate and the individual dose rate components 1-foot from the top lid, as a function of distance from the axis of the 100-ton HI-TRAC, are shown in Figure 5.1.8. For both lids (transfer and top), the reduction in dose rate with increased distance from the cask axial centerline is substantial.

To reduce the dose rate above the water jacket, a localized temporary shield ring, described in Chapter 8, may be employed on the 125-ton HI-TRACs and on the 100-ton HI-TRAC. This temporary shielding, which is water, essentially extends the water jacket to the top of the HI-TRAC. The effect of the temporary shielding on the side dose rate above the water jacket (in the area around the lifting trunnions and the upper flange) is shown on Figure 5.1.9, which shows the dose profile on the side of the 100-ton HI-TRAC with the temporary shielding installed. For comparison, the total dose rate without temporary shielding installed is also shown on Figure 5.1.9. The results indicate that the temporary shielding reduces the dose rate by approximately a factor of 2 in the area above the water jacket.

To illustrate the reduction in dose rate with distance from the side of the 100-ton HI-TRAC, Figure 5.1.10 shows the total dose rate on the surface and at distances of 1-foot and 1-meter.

Figure 5.1.11 plots the total dose rate at various distances from the bottom of the transfer lid, including distances of 1, 5, 10, and 15 feet. Near the transfer lid, the total dose rate is shown to decrease significantly as a function of distance from the 100-ton HI-TRAC axial centerline. Near the axis of the HI-TRAC, the reduction in dose rate from the 1-foot distance to the 15-foot distance is approximately a factor of 15. The dose rate beyond the radial edge of the HI-TRAC is also shown to be relatively low at all distances from the HI-TRAC transfer lid. Thus, prudent transfer operating procedures will employ the use of distance to reduce personnel exposure. In addition, when the HI-TRAC is in the horizontal position and is being transported on site, a missile shield may be positioned in front of the HI-TRAC transfer lid or pool lid. If present, this shield would also serve as temporary gamma shielding which would greatly reduce the dose rate in the vicinity of the transfer lid or pool lid. For example, if the missile shield was a 2 inch thick steel plate, the gamma dose rate would be reduced by approximately 90%.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single HI-STORM cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. These values were conservatively calculated for a burnup of ~~5247,500~~ 5247,500 MWD/MTU and a 53-year cooling time. In addition, the annual dose was calculated for ~~a~~ burnups of 45,000 and 52,500 MWD/MTU *with corresponding cooling times of* ~~and a~~ 9 and 5 years *respectively.* ~~cooling time.~~ BPRAs were ~~not~~ included in these dose estimates. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212, as

stated in Chapter 12, "Operating Controls and Limits". The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the cask array configurations provided in Table 5.1.9. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis fuels. Since the source strengths of the GE 6x6 intact and damaged fuel and the GE 6x6 MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the dose rates from the GE 6x6 fuels for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either GE 6x6 intact or damaged or GE 6x6 MOX fuel for normal conditions is required to demonstrate that the MPC-68 with GE 6x6 fuels will meet the normal condition regulatory requirements. Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E, MPC-32 and the MPC-68.

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM 100 System. Section 5.4.6 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 lists the gamma and neutron sources for the design basis stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.4.

The analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.104 limits and ALARA practices.

### 5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem.



The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

Design basis accidents which may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Scaling this accident dose rate by the dose rate reduction seen in HI-STORM yields a dose rate at the 100 meter controlled area boundary that would be approximately  $1.484.07^\dagger$  mrem/hr for the HI-TRAC accident condition. At this dose rate, it would take  $33781228$  hours ( $\sim 14151$  days) for the dose at the controlled area boundary to reach 5 Rem. *Assuming a 30 day accident duration, the accumulated dose at the controlled area boundary would be 2.93 Rem.* Based on this dose rate and the short duration of use for the

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<sup>†</sup>  $2098.545927.95$  mrem/hr (Table 5.1.10) x [ $129306.6$  mrem/yr (Table 5.4.7) / 8760 hrs /  $20.950.95$  mrem/hr (Table 5.1.5)]

loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

The consequences of the design basis accident conditions for the MPC-68 and MPC-24E storing damaged fuel and the MPC-68F, MPC-68FF, or MPC-24EF storing damaged fuel and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. Analyses in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition does not significantly affect the dose rates around the cask. Therefore, the damaged fuel post-accident dose rates are bounded by the intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.106 limits.

Table 5.1.1

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK  
FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
4535,000 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	15.16	18.14	3.44	36.75	37.68
2	84.79 <sup>†††</sup>	0.05	1.02	85.86	92.07
3	15.88	18.95	2.71	37.54	45.75
4	3.22	1.18	0.95	5.36	6.10
4a	7.12	10.46	13.26	30.83	35.71

<sup>†</sup> Refer to Figure 5.1.12.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> The cobalt activation of incore grid spacers accounts for ~~8.54.1~~ % of this dose rate.

Table 5.1.2

DOSE RATES ADJACENT TO HI-STORM 100 OVERPACK  
FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
5247,500 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	11.14	6.61	3.70	21.46	21.84
2	88.86 <sup>†††</sup>	0.04	2.52	91.41	96.85
3	7.51	4.36	1.84	13.71	15.38
4	1.74	0.49	4.82	7.05	7.51

<sup>†</sup> Refer to Figure 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> The cobalt activation of incore grid spacers accounts for ~~8.04~~ % of this dose rate.

Table 5.1.3

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK FOR NORMAL  
 CONDITIONS  
 MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
 BURNUP AND COOLING TIME  
~~47,500~~40,000 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	15.26	14.43	5.79	35.48
2	77.57	0.01	1.76	79.35
3	6.40	18.63	2.58	27.62
4	1.81	1.42	0.94	4.17
4a	1.77	11.45	12.55	25.77

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<sup>†</sup> Refer to Figure 5.1.12.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.4

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK  
 FOR NORMAL CONDITIONS  
 MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
 BURNUP AND COOLING TIME  
 4535,000 MWD/MTU AND 53-YEAR COOLING

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	10.50	6.08	0.50	17.07	17.89
2	44.21 <sup>†††</sup>	0.39	0.43	45.02	48.25
3	8.31	5.33	0.44	14.08	16.77
4	0.83	0.37	0.42	1.62	1.83

<sup>†</sup> Refer to Figure 5.1.12.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> The cobalt activation of incore grid spacers accounts for ~~8.64.1~~ 8.64.1 % of this dose rate.

Table 5.1.5

DOSE RATES AT ONE METER FROM HI-STORM 100 OVERPACK  
 FOR NORMAL CONDITIONS  
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
 BURNUP AND COOLING TIME  
 5247,500 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	11.15	3.94	0.72	15.82	16.36
2	46.78 <sup>†††</sup>	0.33	1.04	48.16	50.95
3	6.51	2.84	0.28	9.64	10.87
4	0.84	0.22	1.47	2.53	2.66

<sup>†</sup> Refer to Figure 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> The cobalt activation of incore grid spacers accounts for ~~8.04~~ % of this dose rate.

Table 5.1.6

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK  
 FOR NORMAL CONDITIONS  
 MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
 BURNUP AND COOLING TIME  
~~47,500~~ 40,000 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas<sup>††</sup> (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	10.65	4.55	0.77	15.96
2	39.27	0.33	0.74	40.34
3	4.02	5.70	0.42	10.14
4	0.45	0.44	0.40	1.28

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<sup>†</sup> Refer to Figure 5.1.12.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.



Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
42,500/46,000 MWD/MTU AND 53-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	106.76	17.29	849.14	244.86	1218.05	1226.59
2	2673.26 <sup>†</sup>	70.39	0.85	129.91	2874.41	3121.64
3	31.55	3.39	468.20	204.87	708.01	856.53
3 (temp)	14.08	6.03	217.01	3.29	240.41	308.56
4	67.59	1.34	376.81	252.20	697.94	822.44
4 (outer)	20.45	0.85	93.82	170.24	285.36	316.69
5 (pool lid)	704.26	22.94	4298.12	1518.06	6543.38	6608.15
5 (transfer)	1015.91	1.35	6375.30	941.78	8334.34	8431.18
5(t-outer)	262.72	0.46	617.08	372.07	1252.32	1273.80
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	354.02	9.30	126.22	39.80	529.34	561.82
2	1170.82 <sup>†</sup>	21.52	9.99	48.71	1251.03	1360.51
3	148.77	5.18	104.85	19.11	277.92	327.35
3 (temp)	147.95	5.56	89.31	7.23	250.05	294.61
4	23.46	0.23	116.33	62.83	202.86	241.43
5 (transfer)	453.62	0.25	2604.33	262.81	3321.01	3360.14
5(t-outer)	62.33	0.80	234.75	75.45	373.34	377.23

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

<sup>†</sup> The cobalt activation of incore grid spacers accounts for ~~42.36.3%~~ of the surface and one-meter dose rates.

Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRACS FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
~~57,500~~ 75,000 MWD/MTU AND 125-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 125-TON HI-TRACs</b>						
1	6.32	61.85	100.63	415.90	584.70	585.42
2	113.33 <sup>†</sup>	183.20	0.01	287.94	584.49	600.36
3	1.41	6.55	62.26	663.65	733.88	753.59
4	41.57	8.40	340.67	767.94	1158.58	1274.01
4 (outer)	4.84	6.00	42.31	16.11	69.26	83.45
5 (pool)	54.77	3.67	454.56	2883.53	3396.53	3404.24
5 (transfer)	65.81	4.78	601.40	440.29	1112.28	1117.76
<b>ONE METER FROM THE 125-TON HI-TRACs</b>						
1	14.93	24.68	12.90	68.44	120.95	122.99
2	50.47 <sup>†</sup>	59.39	0.52	98.23	208.61	215.68
3	5.66	13.95	12.58	61.07	93.26	98.17
4	11.54	2.03	82.02	79.09	174.68	202.33
5 (transfer)	25.98	0.92	290.76	76.26	393.92	396.85

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

<sup>†</sup> The cobalt activation of incore grid spacers accounts for ~~15.59~~ 4% of the surface and one-meter dose rates.

Table 5.1.9

DOSE RATES FOR ARRAYS OF MPC-24  
WITH DESIGN BASIS ZIRCALOY CLAD FUEL  
AT VARYING BURNUP AND COOLING TIMES

Array Configuration	1 cask	2x2	2x3	2x4	2x5
<b>47,500 MWD/MTU AND 3-YEAR COOLING</b>					
<i>Annual Dose (mrem/year)<sup>†</sup></i>	24.10	18.07	15.86	21.15	16.29
<i>Distance to Controlled Area Boundary (meters)<sup>††,†††</sup></i>	250	350	400	400	450
<b>52,500 MWD/MTU AND 5-YEAR COOLING</b>					
<i>Annual Dose (mrem/year)<sup>†</sup></i>	22.88	14.34	21.52	16.79	20.99
<i>Distance to Controlled Area Boundary (meters)<sup>††</sup></i>	200	300	300	350	350
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
<i>Annual Dose (mrem/year)<sup>†</sup></i>	22.20	23.41	16.77	22.36	14.91
<i>Distance to Controlled Area Boundary (meters)<sup>††</sup></i>	150	200	250	250	300

<sup>†</sup> 8760 hr. annual occupancy is assumed.

<sup>††</sup> Dose location is at the center of the long side of the array.

<sup>†††</sup> Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 53-year cooling, as specified in the ~~Appendix B to the CoC~~ Section 2.1.9, is lower than the burnup used for this analysis.

Table 5.1.10

DOSE RATES AT ONE METER FROM HI-TRAC  
 FOR ACCIDENT CONDITIONS  
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
 AT BOUNDING BURNUP AND COOLING TIMES

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>125-TON HI-TRACs</b>					
<b>57,50075,000 MWD/MTU AND 125-YEAR COOLING</b>					
2 (Accident Condition)	92.26	1.02	3476.98	3570.26	3583.16
2 (Normal Condition)	109.86	0.52	98.23	208.61	215.68
<b>100-TON HI-TRAC</b>					
<b>57,50075,000 MWD/MTU AND 125-YEAR COOLING</b>					
2 (Accident Condition)	1354.67	17.88	4359.16	5731.72	5927.95
2 (Normal Condition)	829.09	9.90	168.82	1007.81	1117.29

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

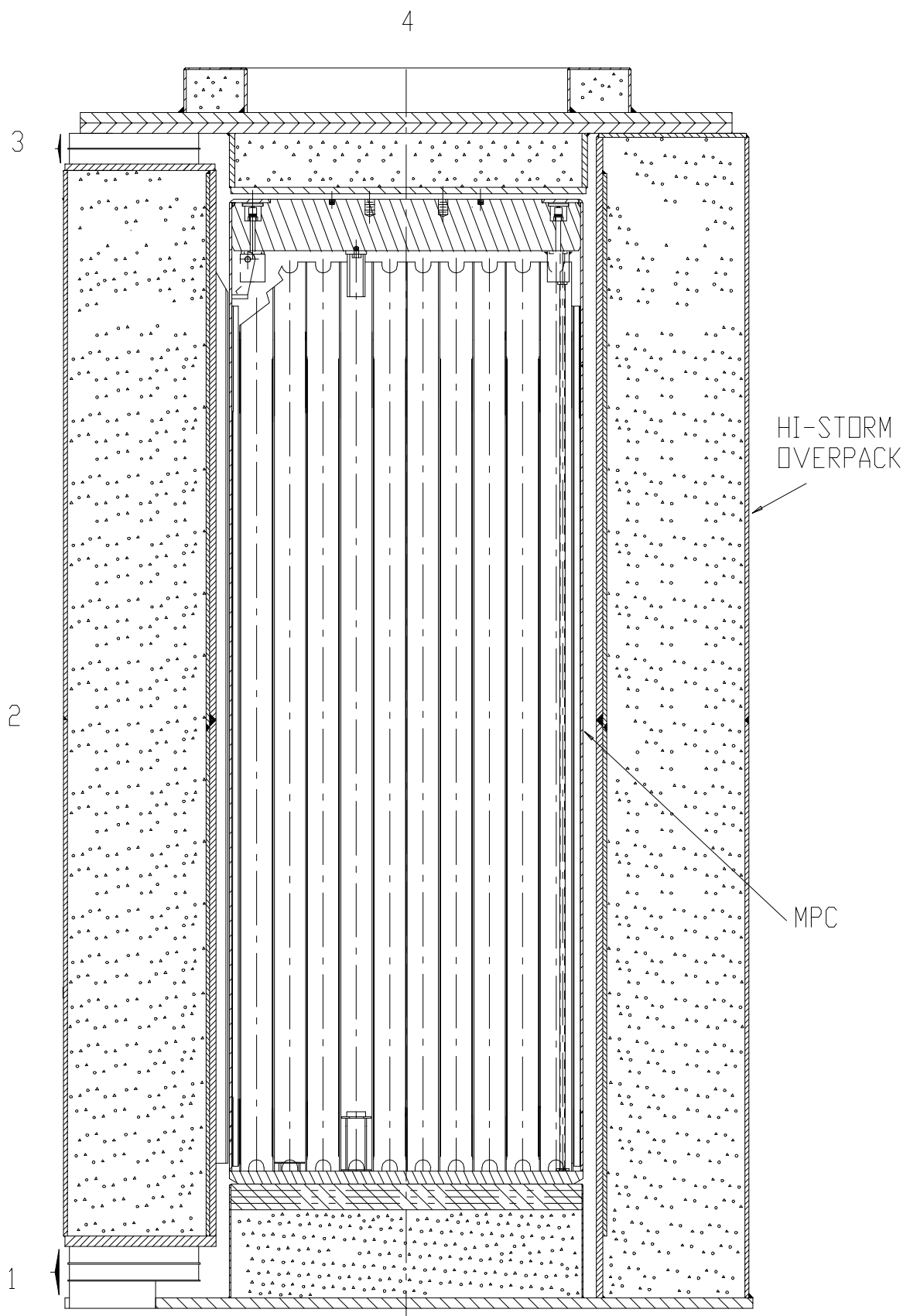


FIGURE 5.1.1; CROSS SECTION ELEVATION VIEW OF HI-STORM 100 OVERPACK WITH DOSE POINT LOCATION

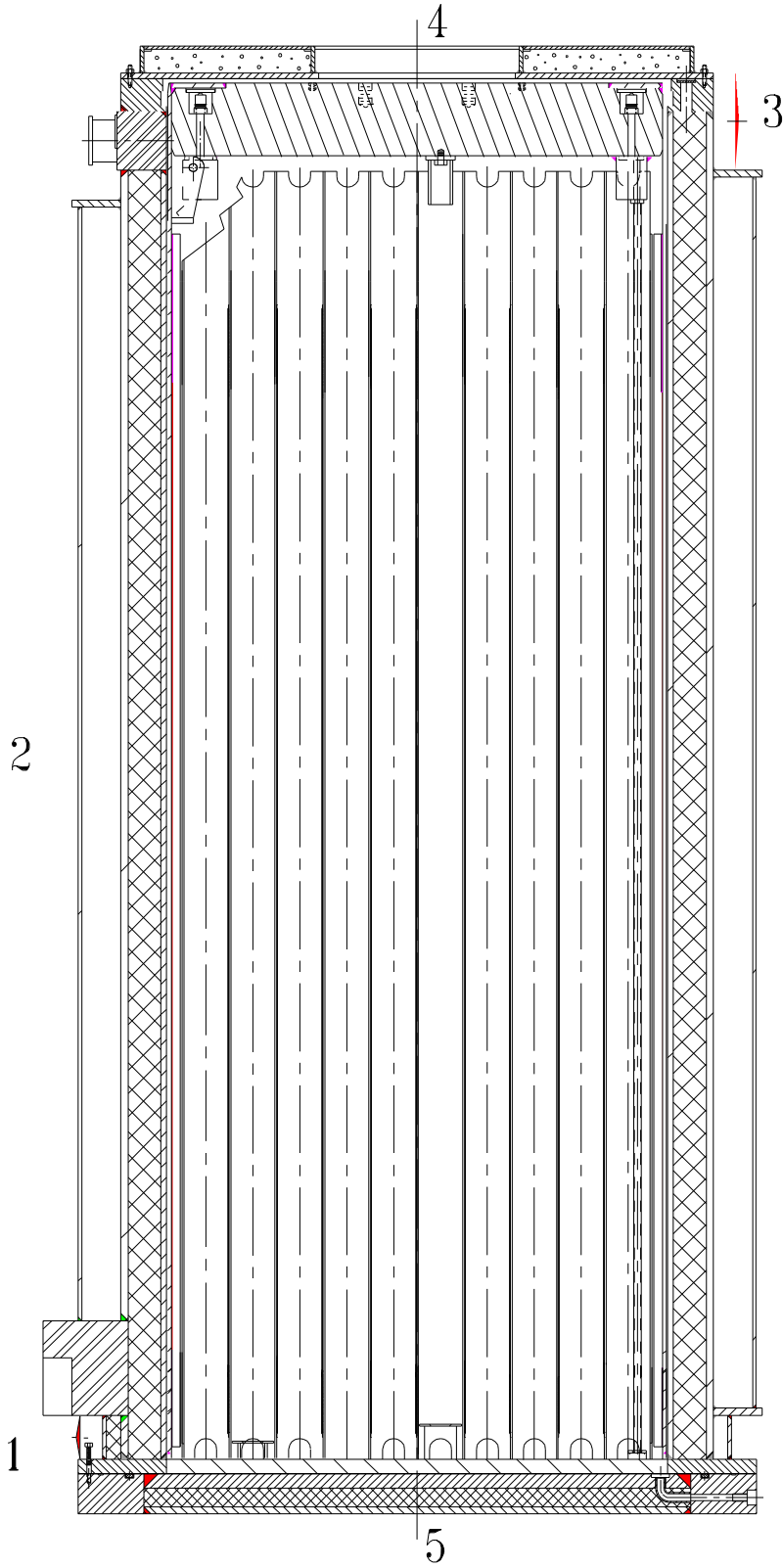


FIGURE 5.1.2; CROSS SECTION ELEVATION VIEW OF 125 TON HI-TRAC TRANSFER CASK WITH DOSE POINT LOCATIONS

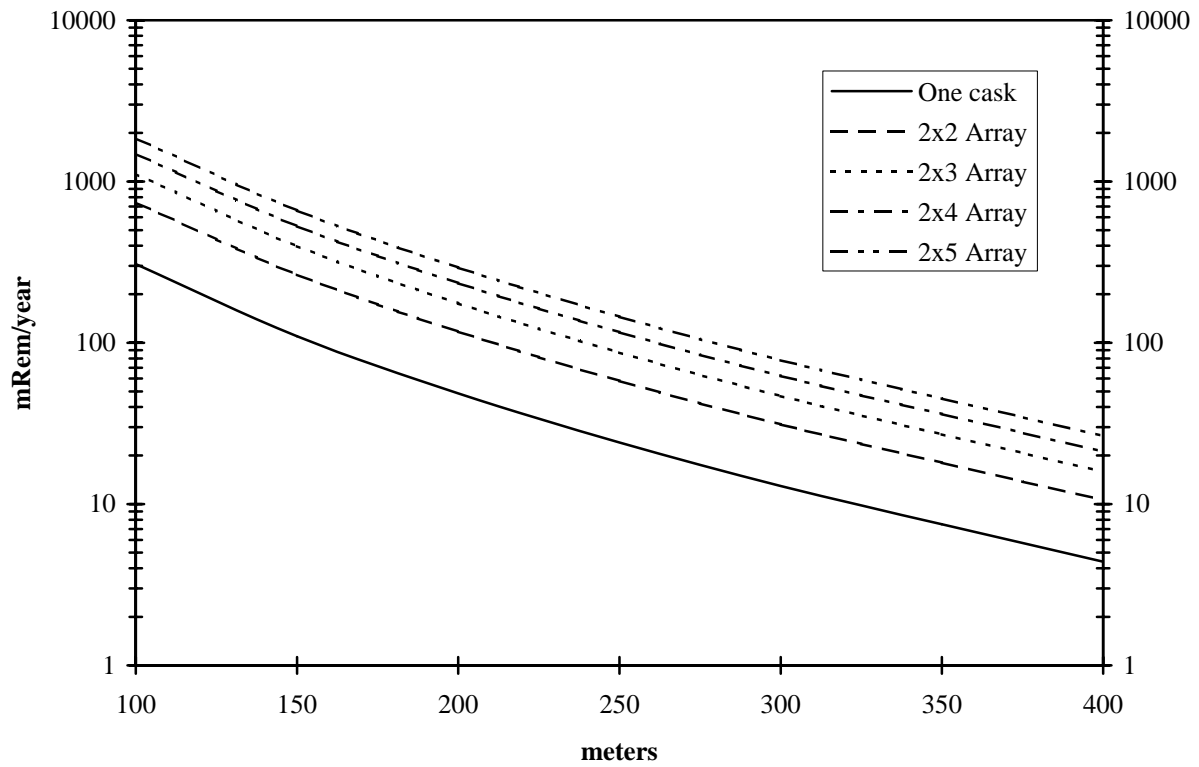


FIGURE 5.1.3; ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 47,500 MWD/MTU AND 3-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

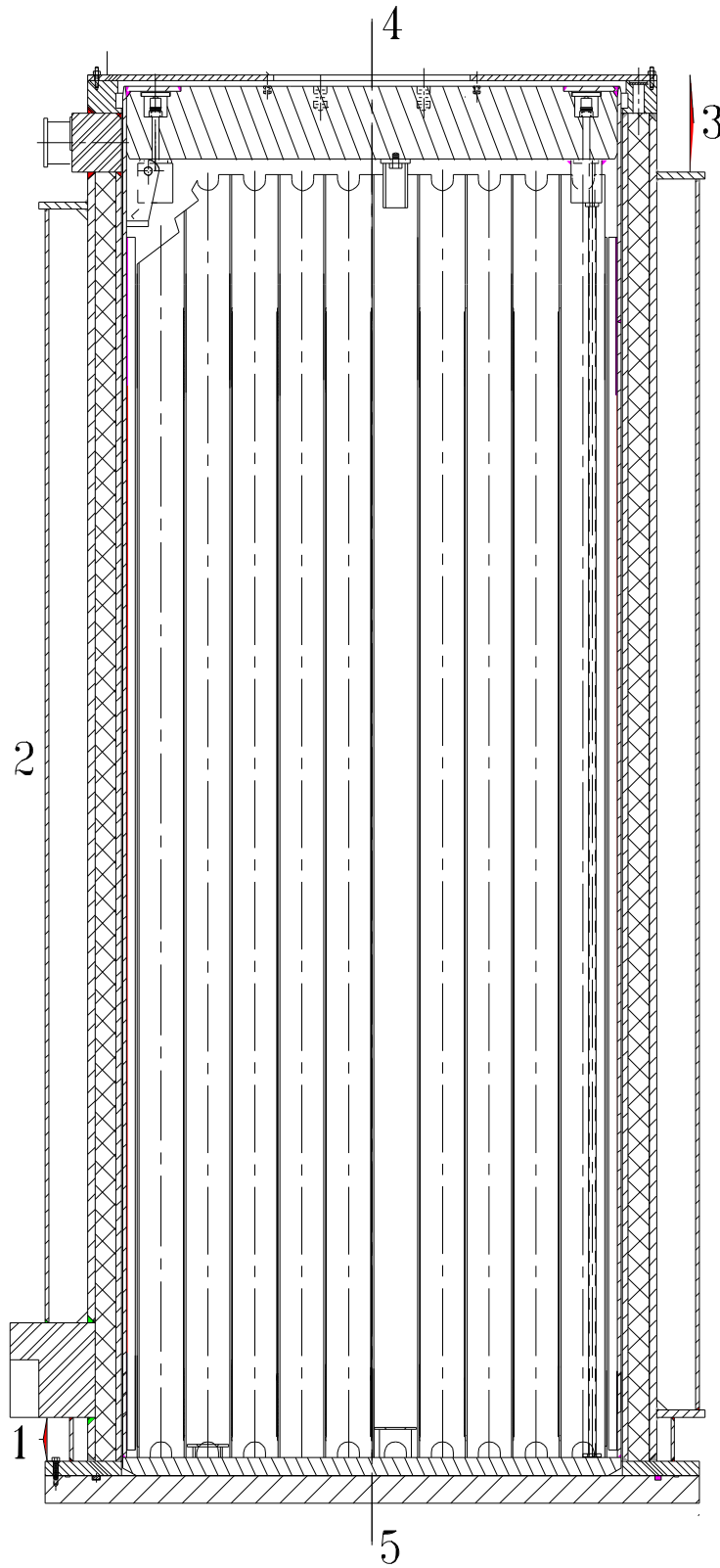


FIGURE 5.1.4; CROSS SECTION ELEVATION VIEW OF 100 TON HI-TRAC TRANSFER CASK (WITH POOL LID) WITH DOSE POINT LOCATIONS



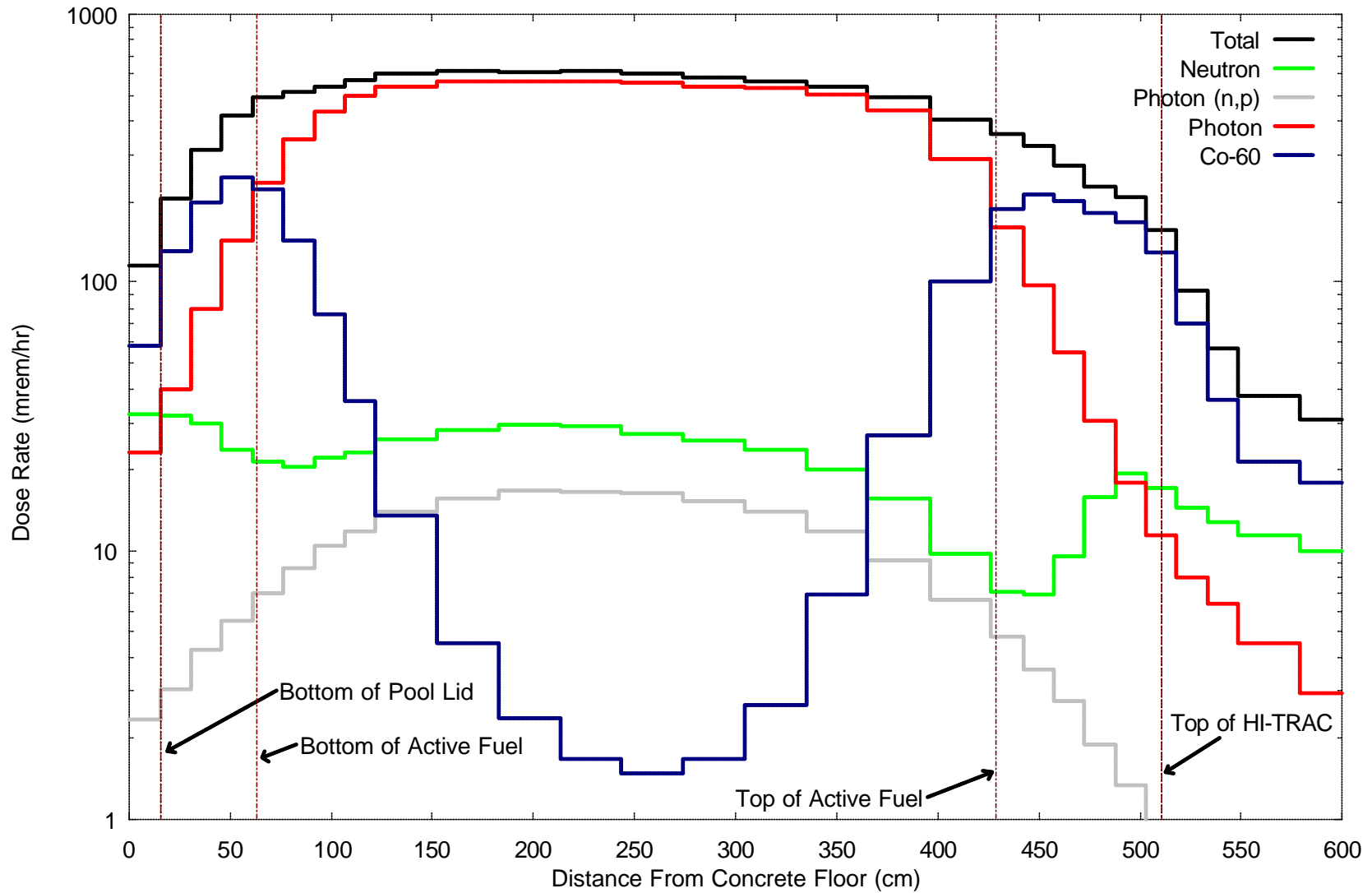


FIGURE 5.1.5; DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

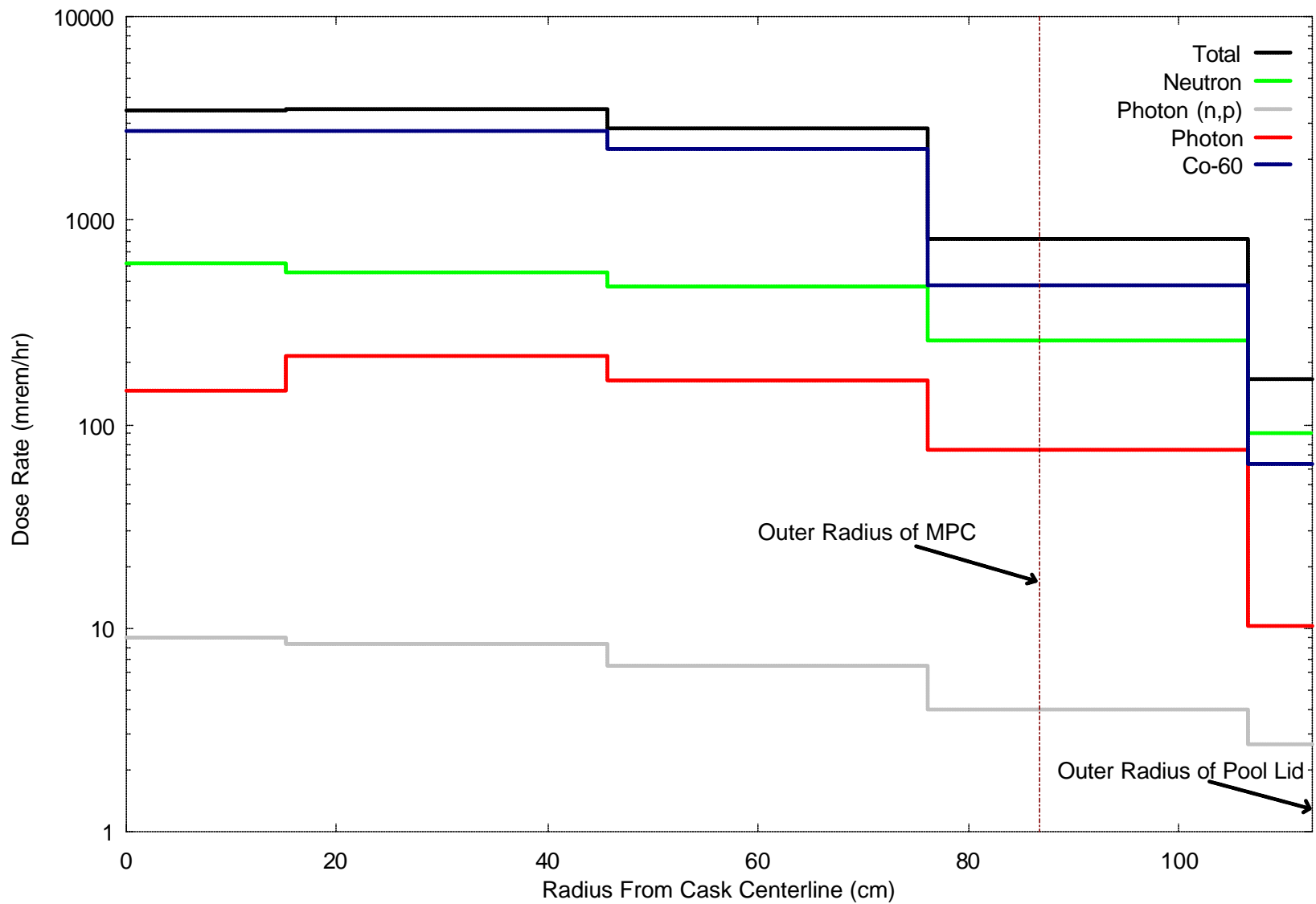


FIGURE 5.1.6; DOSE RATE ON THE SURFACE OF THE POOL LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

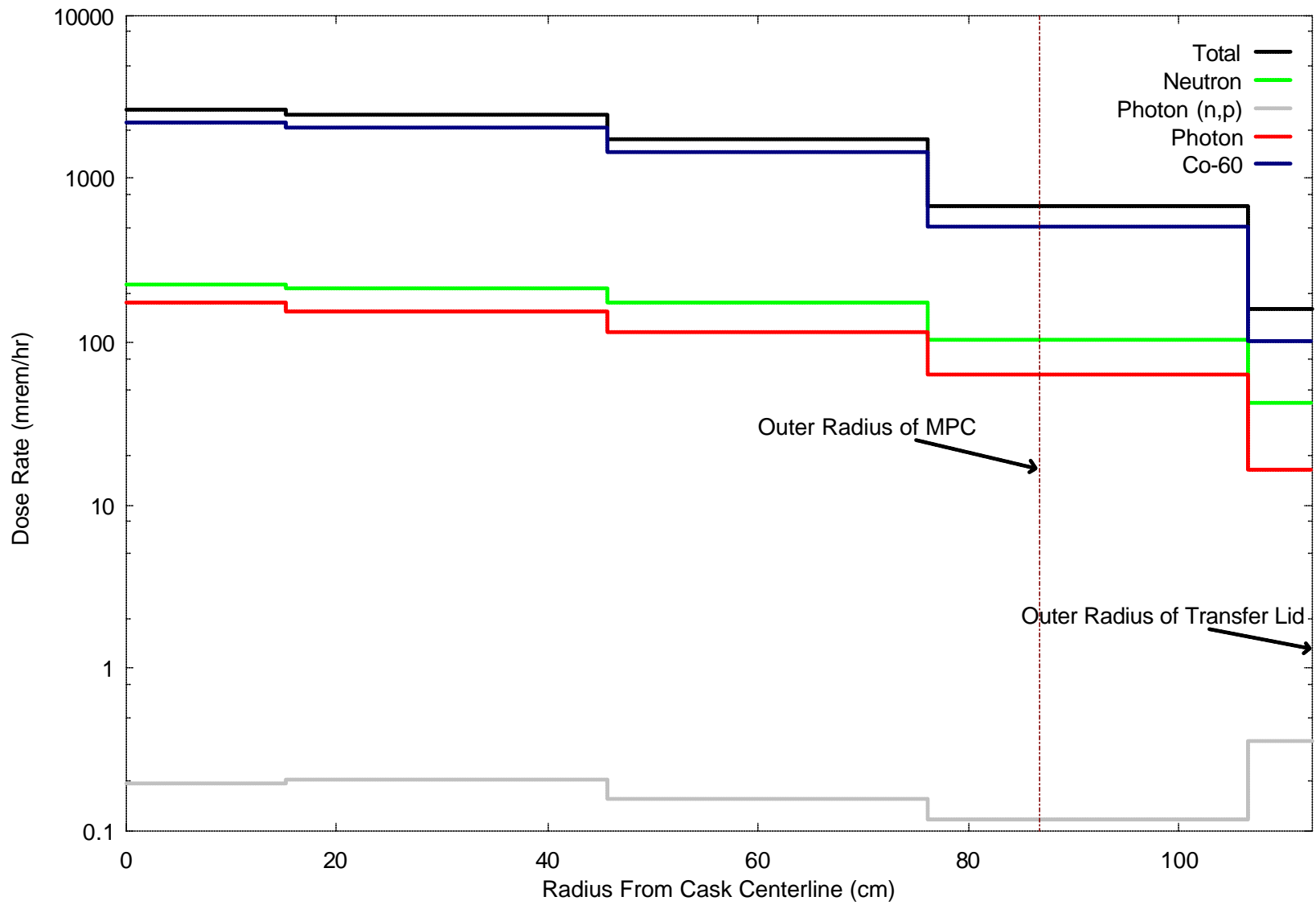


FIGURE 5.1.7; DOSE RATE 1-FOOT FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

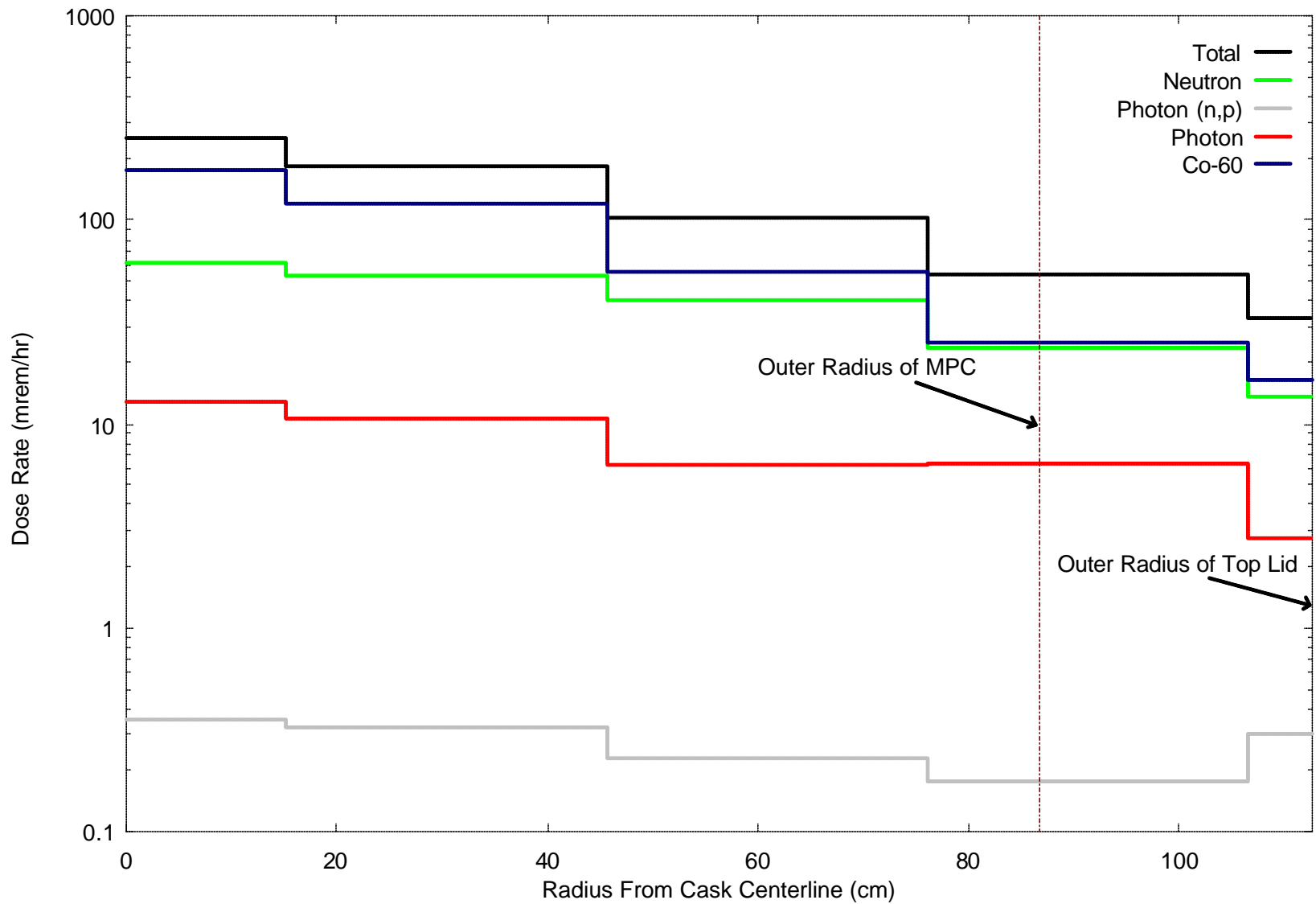


FIGURE 5.1.8; DOSE RATE 1-FOOT FROM THE TOP OF TOP LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

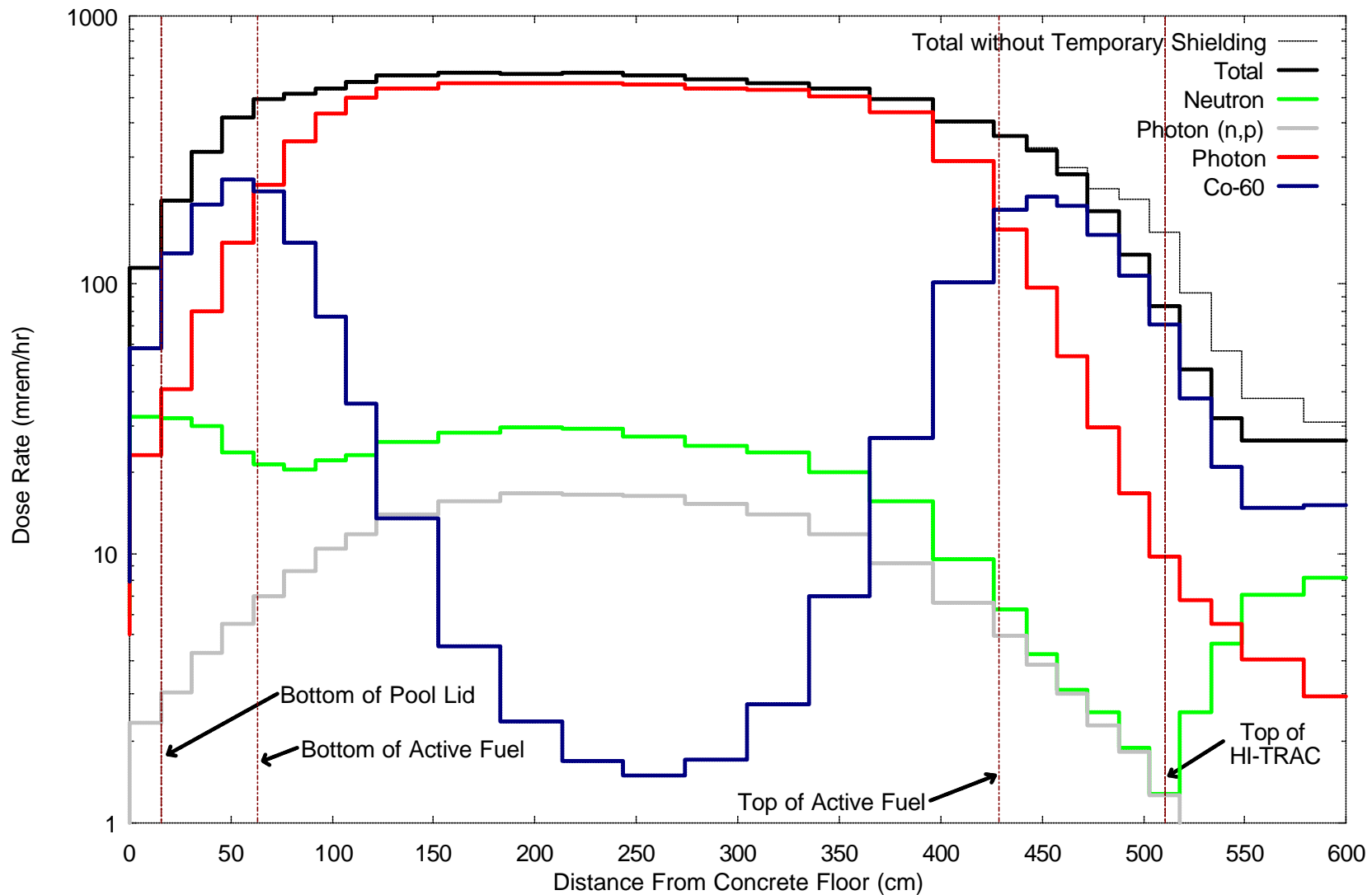


FIGURE 5.1.9; DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH TEMPORARY SHIELDING INSTALLED, WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING (TOTAL DOSE WITHOUT TEMPORARY SHIELDING SHOWN FOR COMPARISON)

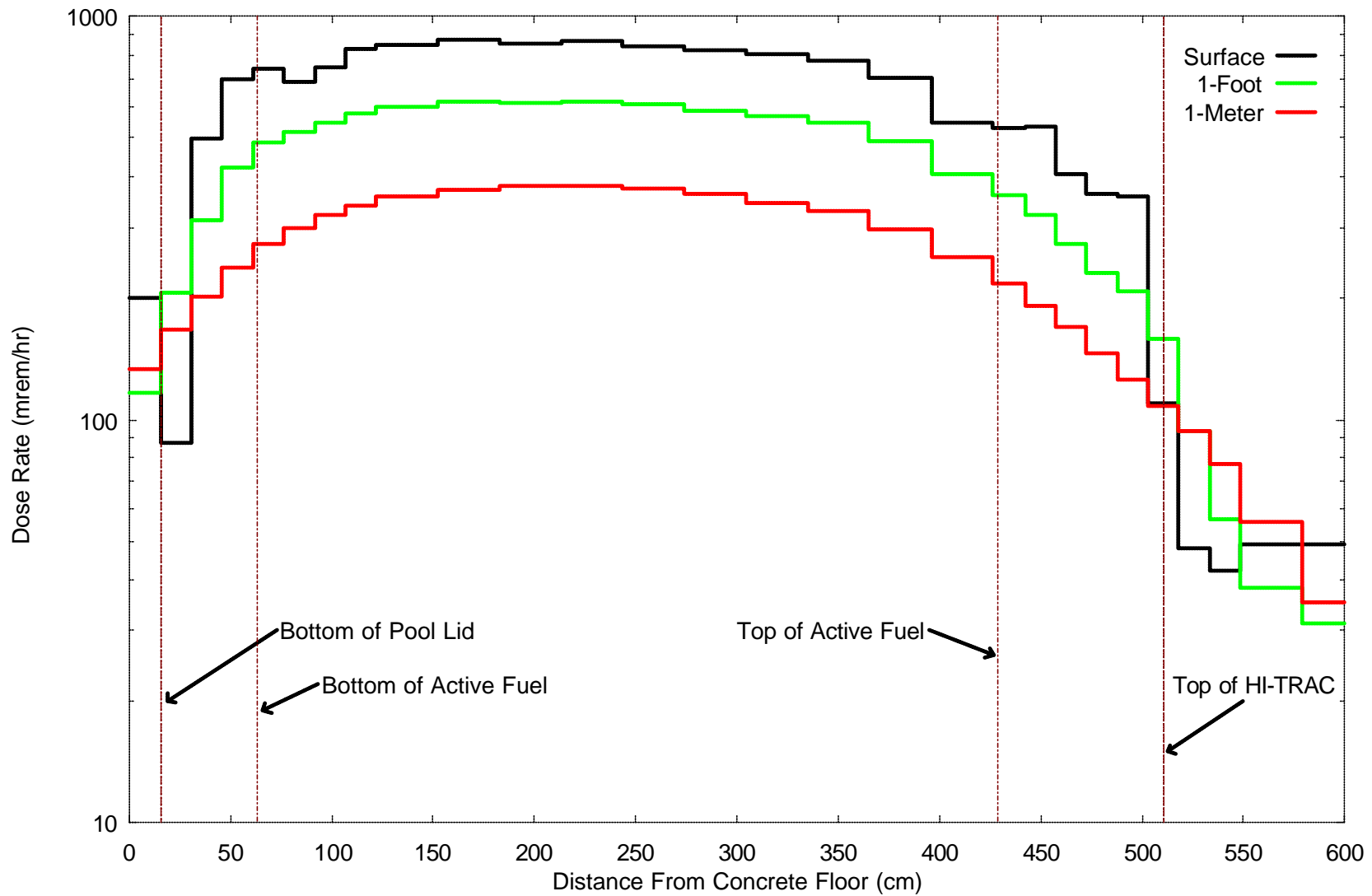


FIGURE 5.1.10; DOSE RATE AT VARIOUS DISTANCES FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

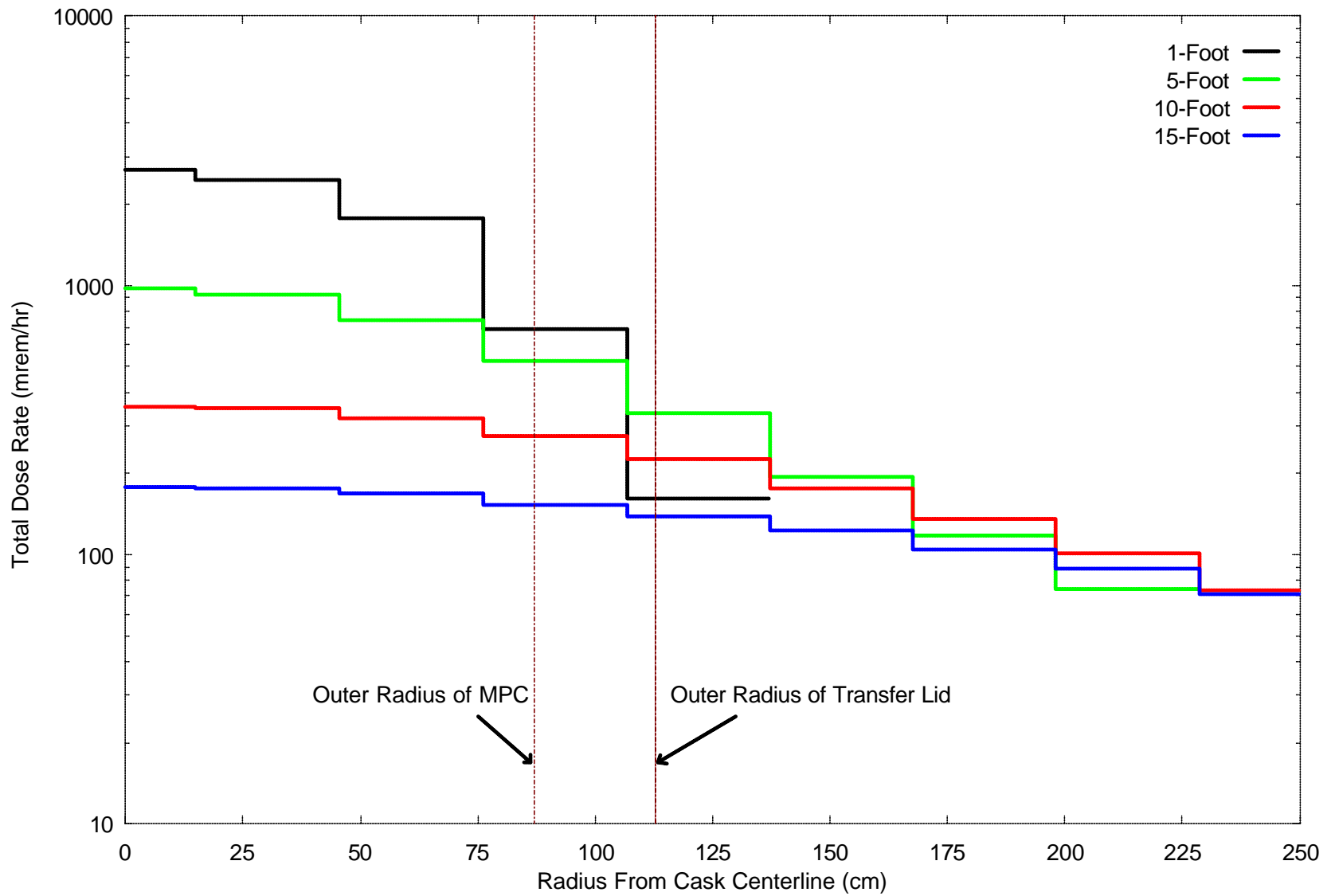


FIGURE 5.1.11; DOSE RATE AT VARIOUS DISTANCES FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

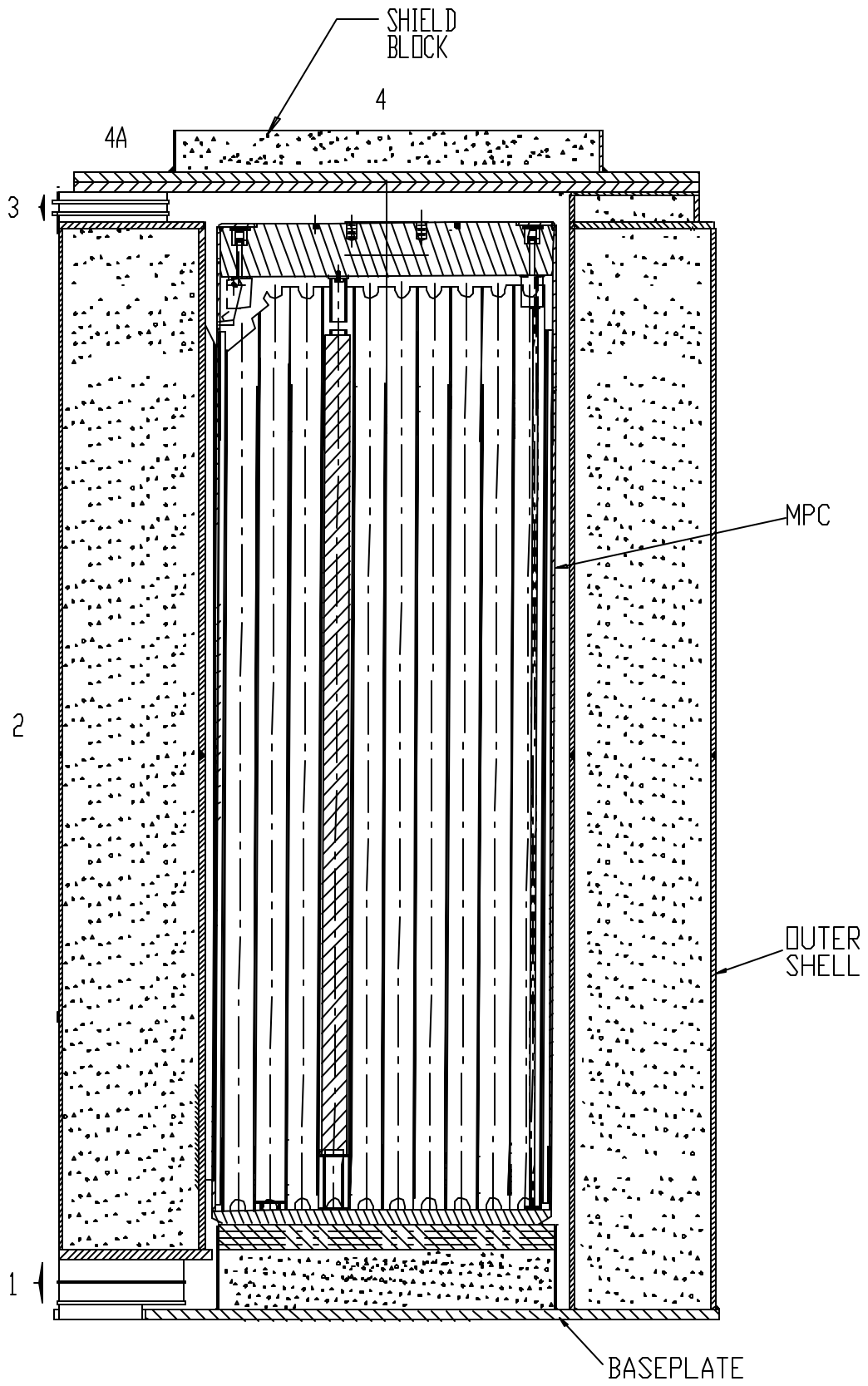


FIGURE 5.1.12; CROSS SECTION ELEVATION VIEW OF THE HI-STORM 100S OVERPACK WITH DOSE POINT LOCATION



## 5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decay heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

Sample input files for SAS2H and ORIGEN-S are provided in Appendices 5.A and 5.B, respectively. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from  $^{60}\text{Co}$  activity of the steel structural material in the fuel element above and below the active fuel region. The third source is from (n, $\gamma$ ) reactions described below.

A description of the design basis zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. The PWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load *for a given burnup and cooling time* from the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun. The BWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load *for a given burnup and cooling time* from the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest  $\text{UO}_2$  mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly is described in Table 5.2.2. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.21 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6 and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger  $^{60}\text{Co}$  activity.

The design basis stainless steel clad fuel assembly for the Indian Point 1, Haddam Neck and San Onofre 1 assembly classes is described in Table 5.2.3. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

The design basis assemblies mentioned above are the design basis assemblies for both intact and damaged fuel and fuel debris for their respective array classes. Analyses of damaged fuel is presented in Section 5.4.2.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

### 5.2.1 Gamma Source

Tables 5.2.4 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuels at varying burnups and cooling times. Tables 5.2.7 and 5.2.22 provides the gamma source in MeV/s and photons/s for the design basis 6x6 and MOX fuel, respectively.

Specific analysis for the HI-STORM 100 System, which includes the HI-STORM storage overpacks and the HI-TRAC transfer casks, was performed to determine the dose contribution from gammas as a function of energy. This analysis considered dose locations external to the 100-ton HI-TRAC transfer cask and the HI-STORM 100 overpack and vents. The results of this analysis have revealed that, due to the magnitude of the gamma source at lower energies, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose at all high dose locations). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Dose rate contributions from above and below this range were evaluated and found to be negligible. Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of  $^{59}\text{Co}$  to  $^{60}\text{Co}$ . The primary source of  $^{59}\text{Co}$  in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant  $^{59}\text{Co}$  impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Conservatively, the impurity level of  $^{59}\text{Co}$  was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

Holtec International has gathered information from utilities and vendors which shows that the 1.0 gm/kg impurity level is very conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. The typical Cobalt-59 impurity level for fuel since the late 1980s is less than 0.5 gm/kg. Based on this, fuel with a short cooling time, 5 to 9 years, would have a Cobalt-59 impurity level less than 0.5 gm/kg. Therefore, the use of a bounding Cobalt-59 impurity level of 1.0 gm/kg is very conservative, particularly for recently manufactured assemblies. Analysis in Reference [5.2.3] indicates that the cobalt impurity in steel and inconel for fuel manufactured in the 1970s ranged from approximately 0.2 gm/kg to 2.2 gm/kg. However, older fuel manufactured with higher cobalt impurity levels will also have a corresponding longer cooling time and therefore will be bounded by the analysis presented in this chapter. As confirmation of this statement, Appendix D presents a comparison of the dose rates around the 100-ton HI-TRAC and the HI-STORM with the MPC-24 for a short cooling time (5 years) using the 1.0 gm/kg mentioned above and for a long cooling time (9 years) using a higher cobalt impurity level of 4.7 gm/kg for inconel. These results confirm that the dose rates for the longer cooling time with the higher impurity level are essentially equivalent to (within 11%) or bounded by the dose rates for the shorter cooling time with the lower impurity level. Therefore, the analysis in this chapter is conservative.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM 100 system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 1 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. The masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative <sup>59</sup>Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a <sup>59</sup>Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a <sup>60</sup>Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the  $^{60}\text{Co}$  is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.
2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.10. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.11 through 5.2.13 provide the  $^{60}\text{Co}$  activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the MPC-32, MPC-24, and the MPC-68 for varying burnup and cooling times. The design basis 6x6 and MOX fuel assemblies are conservatively assumed to have the same  $^{60}\text{Co}$  source strength as the BWR design basis fuel. This is a conservative assumption as the design basis 6x6 fuel and MOX fuel assemblies are limited to a significantly lower burnup and longer cooling time than the design basis fuel.

In addition to the two sources already mentioned, a third source arises from (n, $\gamma$ ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

*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. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cs-134 and Eu-154, two of the major contributors to the gamma source, range from 0.79 to 1.009 and 0.79 to 0.98, respectively. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.*

## 5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the  $^{235}\text{U}$  initial enrichments for various burnup ranges from 20,000 - 70,75,000 MWD/MTU for PWR and 20,000 - 70,000 MWD/MTU for BWR zircaloy clad fuel. These enrichments are based on References [5.2.6] and [5.2.7]. Table 8 of reference [5.2.6] presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, for burnups up to 50,000 MWD/MTU, are approximately the average enrichments from Table 8 of reference [5.2.6] for the burnup range that is 5,000 MWD/MTU less than the ranges listed in

Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] *and some additional data from nuclear plants* was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the *shielding* analysis are adequate to bound the fuel authorized by the limits in the CoC Section 2.1.9 for loading in the HI-STORM system. ~~Therefore a minimum enrichment is not specified in the limits in the CoC.~~ Since the enrichment does affect the source term evaluation, it is recommended that the site-specific dose evaluation consider the enrichment for the fuel being stored.

The neutron source calculated for the design basis fuel assemblies for the MPC-24, MPC-32, and MPC-68 and the design basis 6x6 fuel are listed in Tables 5.2.15 through 5.2.18 in neutrons/s for varying burnup and cooling times. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. <sup>244</sup>Cm accounts for approximately 96-97% of the total number of neutrons produced, ~~with slightly over 2% originating from (α,n) Alpha,n reactions in isotopes other than <sup>244</sup>Cm~~ account for approximately 0.3-2% of the neutrons produced while spontaneous fission in isotopes other than <sup>244</sup>Cm account for approximately 2-8% of the neutrons produced within the UO<sub>2</sub> fuel. ~~The remaining 2% derive from spontaneous fission in various Pu and Cm radionuclides.~~ In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

*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. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cm-244 ranges from 0.81 to 0.95. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.*

### 5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.3 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.3 is actually longer than the true active fuel length of 122 inches for the WE 15x15 and 83 inches for the LaCrosse 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this does not reflect the potential change in dose rates. As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 neutrons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches and 144 inches, respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 neutrons/s/inch and 1 neutron/s/inch for the stainless steel and zircaloy fuel, respectively. The result would be a higher neutron dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.8, 5.2.9, 5.2.19, and 5.2.20 list the gamma and neutron source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV, which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.4 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

### 5.2.4 Non-fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM 100

System as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted to the inner four fuel storage locations in the MPC-24, MPC-24E, and the MPC-32.

#### 5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies. *Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.*

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was

accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRAs which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding BPRAs were actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRAs. The masses of this TPD and BPRAs are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRAs.

Once the bounding BPRAs and TPD were determined, the allowable Co-60 source *and decay heat* from the BPRAs and TPD were specified as: 50 curies Co-60 *and 0.77 watts* for each TPD and ~~834~~895 curies Co-60 *and 14.4 watts* for each BPRAs. Table 5.2.31 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Section 5.4.6 discusses the increase in the cask dose rates due to the insertion of BPRAs or TPDs into fuel assemblies.

#### 5.2.4.2 CRA and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years ( upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B<sub>4</sub>C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in



the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited to four CRAs and/or APSRs. These four devices are required to be stored in the inner four locations in the MPC-24, MPC-24E, MPC-24EF, and MPC-32 as outlined in ~~Appendix B to the CoC~~ *Section 2.1.9*.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

#### Configuration 1: CRA and APSR

This configuration had the lower 15 inches of the CRA and APSR activated at full flux with two regions above the 15 inches activated at a reduced power level. This simulates a CRA or APSR which was operated at 10% insertion. The regions above the 15 inches reflect the upper portion of the fuel assembly.

### Configuration 2: CRA and APSR

This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

### Configuration 3: APSR

This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.

Tables 5.2.34 and 5.2.35 present the source terms, *including decay heat*, that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in ~~Appendix B to the CoC~~ *Section 2.1.9*. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Section 5.4.6 discusses the effect on dose rate of the insertion of APSRs and CRAs into the inner four fuel assemblies in the MPC-24 or MPC-32.

### 5.2.5 Choice of Design Basis Assembly

The analysis presented in this chapter was performed to bound the fuel assembly classes listed in Tables 2.1.1 and 2.1.2. In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly that produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the PWR MPCs (MPC-24 and MPC-32) and the BWR MPCs (MPC-68).

#### 5.2.5.1 PWR Design Basis Assembly

Table 2.1.1 lists the PWR fuel assembly classes that were evaluated to determine the design basis PWR fuel assembly. Within each class, the fuel assembly with the highest UO<sub>2</sub> mass was analyzed. Since the variations of fuel assemblies within a class are very minor (pellet diameter, clad thickness, etc.), it is conservative to choose the assembly with the highest UO<sub>2</sub> mass. For a given class of assemblies, the one with the highest UO<sub>2</sub> mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, the highest UO<sub>2</sub> mass will have produced the most energy and therefore the most fission products.

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR 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 class is the assembly with the highest UO<sub>2</sub> mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Indian Point 1, Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. 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<sub>2</sub> mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO<sub>2</sub> 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 BWR Design Basis Assembly

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<sub>2</sub> 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<sub>2</sub> mass. For a given array type of assemblies, the one with the highest UO<sub>2</sub> 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<sub>2</sub> 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<sub>2</sub> mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO<sub>2</sub> 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 ~~Appendix B to the CoC~~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 ~~Appendix B to the CoC~~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<sub>2</sub> 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<sub>2</sub> 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 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. burnup versus cooling time limits in the CoC that are based on a maximum permissible decay heat per assembly.~~ 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.% <sup>235</sup>U

The coefficients for this equation were developed by fitting ORIGEN-S calculated data for a specific cooling time. ORIGEN-S calculations were performed for enrichments ranging from 0.7 to 5.0 wt.% <sup>235</sup>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. Since the decay heat increases as the enrichment decreases, the allowable burnup will decrease as the enrichment decreases. Therefore, the enrichment used to calculate 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.*

~~The decay heat values per assembly were calculated using the methodology described in Section 5.2. 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 the BWR array classes. A penalty was not applied to the PWR array classes since the thermal analysis in Chapter 4 has more than a 5% margin in the calculated allowable decay heat.*

~~The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time limits in the CoC. The enrichments used in the calculation of the decay heats were consistent with Table 5.2.24. As demonstrated in Tables 5.2.27 and 5.2.28, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, Appendix B to the CoC limits the heavy metal mass to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.~~

~~As further a demonstration that the decay heat values used to determine the allowable burnups (calculated using the design basis fuel assemblies) are conservative, a comparison between these~~

calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. 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].

~~The heavy metal mass of the non design basis fuel assembly classes in Appendix B of the Certificate of Compliance are limited to the masses used in Tables 5.2.25 and 5.2.26. No margin is applied between the allowable mass and the analyzed mass of heavy metal for the non design basis fuel assemblies. This is acceptable because additional assurance that the decay heat values for the non design basis fuel assemblies are bounding values is obtained by using the decay heat values for the design basis fuel assemblies to determine the acceptable storage criteria for all fuel assemblies. As mentioned above, Table 5.2.29 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.~~

As mentioned above, the fuel assembly burnup and cooling times in ~~Appendix B to the CoC~~Section 2.1.9 were calculated using the decay heat limits which are also stipulated in ~~Appendix B to the CoC~~Section 2.1.9. The burnup and cooling times for the non-fuel hardware, in ~~Appendix B to the CoC~~Section 2.1.9, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup, *decay heat*, and *enrichment equations and cooling times* were ~~calculated~~ 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 ~~Appendix B to the CoC~~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.

### 5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister.

Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STORM 100 System.

### 5.2.7 Fuel Assembly Neutron Sources

Neutron sources are used in reactors during initial startup of reactor cores. There are different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STORM 100 System. Currently these are the only neutron source permitted for storage in the HI-STORM 100 System.

### 5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which, have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and ~~Appendix B to the~~ *Section 2.1.9* requires that these channels be stored in the inner sixteen locations.

The activation of a single stainless steel channel was calculated by simulating the irradiation of the channels with ORIGEN-S using the flux calculated from the LaCrosse fuel assembly. The mass of the steel channel in the active fuel zone (83 inches) was used in the analysis. For burnups beyond 22,500 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 22,500 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 22,500 MWD/MTU.

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming



the plant operated continually from 11/69 until 4/87, approximately 17.5 years or 6388 days, the accumulated burnup for the channels would be 186,000 MWD/MTU (6388 days times 29.17 MW/MTU from Table 5.2.3). Therefore, the cobalt activity calculated for a single stainless steel channel irradiated for 180,000 MWD/MTU was calculated to be 667 curies of Co-60 for 13 years cooling. This is equivalent to a source of  $4.94\text{E}+13$  photons/sec in the energy range of 1.0-1.5 MeV.

In order to demonstrate that sixteen stainless steel channels are acceptable for storage in an MPC-68, a comparison of source terms is performed. Table 5.2.8 indicates that the source term for the LaCrosse design basis fuel assembly in the 1.0-1.5 MeV range is  $6.34\text{E}+13$  photons/sec for 10 years cooling, assuming a 144 inch active fuel length. This is equivalent to  $4.31\text{E}+15$  photons/sec/cask. At 13 years cooling, the fuel source term in that energy range decreases to  $4.31\text{E}+13$  photons/sec which is equivalent to  $2.93\text{E}+15$  photons/sec/cask. If the source term from the stainless steel channels is scaled to 144 inches and added to the 13 year fuel source term the result is  $4.30\text{E}+15$  photons/sec/cask ( $2.93\text{E}+15$  photons/sec/cask +  $4.94\text{E}+13$  photons/sec/channel x 144 inch/83 inch x 16 channels/cask). This number is equivalent to the 10 year  $4.31\text{E}+15$  photons/sec/cask source calculated from Table 5.2.8 and used in the shielding analysis in this chapter. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.

Table 5.2.1

## DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	<b>PWR</b>	<b>BWR</b>
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	3.6	3.2
Burnup (MWD/MTU) <sup>†</sup>	<del>52,500 (MPC-24)</del> 45,000 (MPC-32)	47,500 (MPC-68)
Cooling Time (years) <sup>†</sup>	5 (MPC-24 and 32)	5 (MPC-68)
Specific power (MW/MTU)	40	30
Weight of UO <sub>2</sub> (kg) <sup>††</sup>	562.029	225.177
Weight of U (kg) <sup>††</sup>	495.485	198.516

## Notes:

1. The B&W 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun.
2. The GE 7x7 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

<sup>†</sup> Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times listed in Appendix B to the CoC Section 2.1.9.

<sup>††</sup> Derived from parameters in this table.

Table 5.2.1 (continued)

## DESCRIPTION OF DESIGN BASIS FUEL

	<b>PWR</b>	<b>BWR</b>
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2

DESCRIPTION OF DESIGN BASIS GE 6x6 ZIRCALOY CLAD FUEL

	<b>BWR</b>
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	129.5
Weight of U (kg) <sup>†</sup>	114.2

Notes:

1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

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<sup>†</sup> Derived from parameters in this table.

Table 5.2.3

DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

	<b>PWR</b>	<b>BWR</b>
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U)	3.5	3.5
Burnup (MWD/MTU) <sup>†</sup>	40,000 (MPC-24 and 32)	22,500 (MPC-68)
Cooling Time (years) <sup>†</sup>	8 (MPC-24), 9 (MPC-32)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

Notes:

1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: Indian Point 1, Haddam Neck, and San Onofre 1.
2. The LaCrosse 10x10 is the design basis assembly for the following fuel assembly class listed in Table 2.1.2: LaCrosse.

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<sup>†</sup> Burnup and cooling time combinations are equivalent to or conservatively bound the limits in Appendix B to the CoC Section 2.1.9.

Table 5.2.4

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>45,000-35,000 MWD/MTU 5-3 Year Cooling</b>		<b>4575,000 MWD/MTU 10-8 Year Cooling</b>	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	<i>2.30E+15</i>	<i>4.00E+15</i>	<i>2.52E+15</i>	<i>4.39E+15</i>
0.7	1.0	<i>9.62E+14</i>	<i>1.13E+15</i>	<i>5.41E+14</i>	<i>6.36E+14</i>
1.0	1.5	<i>2.18E+14</i>	<i>1.75E+14</i>	<i>1.66E+14</i>	<i>1.33E+14</i>
1.5	2.0	<i>2.45E+13</i>	<i>1.40E+13</i>	<i>7.51E+12</i>	<i>4.29E+12</i>
2.0	2.5	<i>3.57E+13</i>	<i>1.59E+13</i>	<i>6.94E+11</i>	<i>3.08E+11</i>
2.5	3.0	<i>9.59E+11</i>	<i>3.49E+11</i>	<i>4.99E+10</i>	<i>1.81E+10</i>
Total		<i>3.54E+15</i>	<i>5.34E+15</i>	<i>3.24E+15</i>	<i>5.16E+15</i>

Table 5.2.5

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	Upper Energy	<del>42,500</del> 46,000 MWD/MTU 5-3 Year Cooling		5247,500 MWD/MTU 5-3 Year Cooling		<del>57,500</del> 75,000 MWD/MTU 12-5 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	3.14E+15	5.45E+15	3.25E+15	5.65E+15	3.55E+15	6.17E+15
0.7	1.0	1.43E+15	1.68E+15	1.49E+15	1.75E+15	1.36E+15	1.60E+15
1.0	1.5	3.07E+14	2.46E+14	3.17E+14	2.53E+14	2.94E+14	2.35E+14
1.5	2.0	2.97E+13	1.70E+13	3.03E+13	1.73E+13	1.50E+13	8.59E+12
2.0	2.5	3.80E+13	1.69E+13	3.83E+13	1.70E+13	7.63E+12	3.39E+12
2.5	3.0	1.16E+12	4.22E+11	1.19E+12	4.33E+11	3.72E+11	1.35E+11
Total		4.94E+15	7.42E+15	5.12E+15	7.69E+15	5.23E+15	8.02E+15

Table 5.2.6

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	Upper Energy	39,000 MWD/MTU 3 Year Cooling		40,000 MWD/MTU 5-3 Year Cooling		47,500 MWD/MTU 5-6 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	1.00E+15	1.74E+15	1.02E+15	1.78E+15	1.10E+15	1.91E+15
0.7	1.0	4.25E+14	4.99E+14	4.37E+14	5.14E+14	3.21E+14	3.78E+14
1.0	1.5	9.18E+13	7.35E+13	9.40E+13	7.52E+13	7.67E+13	6.13E+13
1.5	2.0	9.19E+12	5.25E+12	9.27E+12	5.30E+12	3.55E+12	2.03E+12
2.0	2.5	1.17E+13	5.18E+12	1.17E+13	5.21E+12	1.03E+12	4.57E+11
2.5	3.0	3.69E+11	1.34E+11	3.70E+11	1.35E+11	5.83E+10	2.12E+10
Total		1.54E+15	2.32E+15	1.58E+15	2.38E+15	1.50E+15	2.35E+15



Table 5.2.7

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>30,000 MWD/MTU 18-Year Cooling</b>	
		(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.53e+14	2.65e+14
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+09	5.99e+08
2.5	3.0	7.30e+07	2.66e+07
Totals		1.61e+14	2.73e+14

Table 5.2.8

CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	Upper Energy	22,500 MWD/MTU 10-Year Cooling	
		(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	2.72e+14	4.74e+14
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		3.72e+14	5.61e+14

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the CoC~~ Section 2.1.9 are based on the actual 83-inch active fuel length.

Table 5.2.9

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	Upper Energy	40,000 MWD/MTU 8-Year Cooling		40,000 MWD/MTU 9-Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.37e+15	2.38e+15	1.28E+15	2.22E+15
7.0e-01	1.0	2.47e+14	2.91e+14	1.86E+14	2.19E+14
1.0	1.5	4.59e+14	3.67e+14	4.02E+14	3.21E+14
1.5	2.0	3.99e+12	2.28e+12	3.46E+12	1.98E+12
2.0	2.5	5.85e+11	2.60e+11	2.69E+11	1.20E+11
2.5	3.0	3.44e+10	1.25e+10	1.77E+10	6.44E+09
Totals		2.08e+15	3.04e+15	1.87E+15	2.76E+15

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the~~ *Section 2.1.9* are based on the actual 122-inch active fuel length.

Table 5.2.10

SCALING FACTORS USED IN CALCULATING THE  $^{60}\text{Co}$  SOURCE

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

Table 5.2.11

CALCULATED MPC-32 <sup>60</sup>Co SOURCE PER ASSEMBLY FOR DESIGN BASIS  
 ZIRCALOY CLAD FUEL  
 AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b>45,00035,000 MWD/MTU and 53-Year Cooling (curies)</b>	<b>4575,000 MWD/MTU and 108-Year Cooling (curies)</b>
Lower End Fitting	184.28	147.77
Gas Plenum Springs	14.06	11.27
Gas Plenum Spacer	8.07	6.47
Expansion Springs	N/A	N/A
Incore Grid Spacers	477.26	382.69
Upper End Fitting	90.39	72.48
Handle	N/A	N/A

Table 5.2.12

CALCULATED MPC-24 <sup>60</sup>Co SOURCE PER ASSEMBLY FOR DESIGN BASIS  
 ZIRCALOY CLAD FUEL  
 AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b><del>42,500</del>46,000 MWD/MTU and 53-Year Cooling (curies)</b>	<b><del>5247,500</del> MWD/MTU and 53-Year Cooling (curies)</b>	<b><del>57,500-75,000</del> MWD/MTU and - 12-5 Year Cooling (curies)</b>
Lower End Fitting	221.36	227.04	219.47
Gas Plenum Springs	16.89	17.32	16.74
Gas Plenum Spacer	9.69	9.94	9.61
Expansion Springs	N/A	N/A	N/A
Incore Grid Spacers	573.30	588.00	568.40
Upper End Fitting	108.58	111.36	107.65
Handle	N/A	N/A	N/A

Table 5.2.13

CALCULATED MPC-68 <sup>60</sup>Co SOURCE PER ASSEMBLY FOR DESIGN BASIS  
 ZIRCALOY CLAD FUEL  
 AT DESIGN BASIS BURNUP AND COOLING TIME

<b>Location</b>	<b><i>39,000 MWD/MTU and 3-Year Cooling (curies)</i></b>	<b><i>40,000 MWD/MTU and 53-Year Cooling (curies)</i></b>	<b><i>47,500-70,000 MWD/MTU and 56-Year Cooling (curies)</i></b>
Lower End Fitting	82.47	82.69	68.73
Gas Plenum Springs	25.20	25.27	21.00
Gas Plenum Spacer	N/A	N/A	N/A
Expansion Springs	4.58	4.59	3.82
Grid Spacer Springs	37.80	37.90	31.50
Upper End Fitting	22.91	22.97	19.09
Handle	2.86	2.87	2.39

Table 5.2.14

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Table 5.2.15

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>45,000 MWD/MTU 53-Year Cooling (Neutrons/s)</b>	<b>4575,000 MWD/MTU 108-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	7.80E+06	5.97E+07
4.0e-01	9.0e-01	3.99E+07	3.05E+08
9.0e-01	1.4	3.65E+07	2.79E+08
1.4	1.85	2.70E+07	2.05E+08
1.85	3.0	4.79E+07	3.61E+08
3.0	6.43	4.33E+07	3.29E+08
6.43	20.0	3.82E+06	2.92E+07
Totals		2.06E+08	1.57E+09

Table 5.2.16

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>42,500 46,000 MWD/MTU 53-Year Cooling (Neutrons/s)</b>	<b>5247,500 MWD/MTU 53-Year Cooling (Neutrons/s)</b>	<b>57,500-75,000 MWD/MTU 125-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	<i>1.96E+07</i>	<i>2.19E+07</i>	<i>6.82E+07</i>
4.0e-01	9.0e-01	<i>1.00E+08</i>	<i>1.12E+08</i>	<i>3.48E+08</i>
9.0e-01	1.4	<i>9.16E+07</i>	<i>1.02E+08</i>	<i>3.18E+08</i>
1.4	1.85	<i>6.75E+07</i>	<i>7.54E+07</i>	<i>2.34E+08</i>
1.85	3.0	<i>1.19E+08</i>	<i>1.33E+08</i>	<i>4.11E+08</i>
3.0	6.43	<i>1.08E+08</i>	<i>1.21E+08</i>	<i>3.75E+08</i>
6.43	20.0	<i>9.60E+06</i>	<i>1.07E+07</i>	<i>3.34E+07</i>
Totals		<i>5.16E+08</i>	<i>5.76E+08</i>	<i>1.79E+09</i>

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL  
 FOR VARYING BURNUPS AND COOLING TIMES

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>39,000 MWD/MTU 3-Year Cooling (Neutrons/s)</b>	<b>40,000 MWD/MTU 53-Year Cooling (Neutrons/s)</b>	<b>47,50070,000 MWD/MTU 56-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	5.22E+06	5.45E+06	1.98E+07
4.0e-01	9.0e-01	2.67E+07	2.78E+07	1.01E+08
9.0e-01	1.4	2.44E+07	2.55E+07	9.26E+07
1.4	1.85	1.80E+07	1.88E+07	6.81E+07
1.85	3.0	3.18E+07	3.32E+07	1.20E+08
3.0	6.43	2.89E+07	3.02E+07	1.09E+08
6.43	20.0	2.56E+06	2.67E+06	9.71E+06
Totals		1.37E+08	1.44E+08	5.20E+08

Table 5.2.18

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>30,000 MWD/MTU 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Total		2.20e+7

Table 5.2.19

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	22,500 MWD/MTU 10-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the~~ ~~CoC~~ ~~Section 2.1.9~~ are based on the actual 83-inch active fuel length.

Table 5.2.20

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY  
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 8-Year Cooling (Neutrons/s)	40,000 MWD/MTU 9-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.04e+7	1.01E+07
4.0e-01	9.0e-01	5.33e+7	5.14E+07
9.0e-01	1.4	4.89e+7	4.71E+07
1.4	1.85	3.61e+7	3.48E+07
1.85	3.0	6.41e+7	6.18E+07
3.0	6.43	5.79e+7	5.58E+07
6.43	20.0	5.11e+6	4.92E+06
Totals		2.76e+8	2.66E+08

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the~~ ~~CoC~~ *Section 2.1.9* are based on the actual 122-inch active fuel length.

Table 5.2.21

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	<b>BWR</b>
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO <sub>2</sub> and PuUO <sub>2</sub>
No. of UO <sub>2</sub> Rods	27
No. of PuUO <sub>2</sub> rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o <sup>235</sup> U) <sup>†</sup>	2.24 (UO <sub>2</sub> rods) 0.711 (PuUO <sub>2</sub> rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO <sub>2</sub> ,PuUO <sub>2</sub> (kg) <sup>††</sup>	123.3
Weight of U,Pu (kg) <sup>††</sup>	108.7

<sup>†</sup> See Table 5.3.3 for detailed composition of PuUO<sub>2</sub> rods.

<sup>††</sup> Derived from parameters in this table.

Table 5.2.22

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

<b>Lower Energy</b>	<b>Upper Energy</b>	<b>30,000 MWD/MTU 18-Year Cooling</b>	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.45e+14	2.52e+14
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		1.53e+14	2.60e+14



Table 5.2.23

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY  
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>30,000 MWD/MTU 18-Year Cooling (Neutrons/s)</b>
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.24

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	Initial Enrichment (wt.% <sup>235</sup> U)
<b>BWR Fuel</b>	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	3.6
55,000-60,000	4.0
60,000-65,000	4.4
65,000-70,000	4.8
<b>PWR Fuel</b>	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	3.9
55,000-60,000	4.2
60,000-65,000	4.5
65,000-70,000	4.8
70,000-75,000	5.0

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. *This note does not apply to the maximum burnups of 70,000 and 75,000 MWD/MTU.*

Table 5.2.25 (page 1 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	WE 14×14	WE 14x14	WE 15×15	WE 17×17	WE 17x17
<i>Fuel assembly array class</i>	<i>14x14B</i>	<i>14x14A</i>	<i>15x15ABC</i>	<i>17x17B</i>	<i>17x17A</i>
Active fuel length (in.)	144	<i>144</i>	144	144	<i>144</i>
No. of fuel rods	179	<i>179</i>	204	264	<i>264</i>
Rod pitch (in.)	0.556	<i>0.556</i>	0.563	0.496	<i>0.496</i>
Cladding material	Zr-4	<i>Zr-4</i>	Zr-4	Zr-4	<i>Zr-4</i>
Rod diameter (in.)	0.422	<i>0.4</i>	0.422	0.374	<i>0.36</i>
Cladding thickness (in.)	0.0243	<i>0.0243</i>	0.0245	0.0225	<i>0.0225</i>
Pellet diameter (in.)	0.3659	<i>0.3444</i>	0.3671	0.3232	<i>0.3088</i>
Pellet material	UO <sub>2</sub>	<i>UO<sub>2</sub></i>	UO <sub>2</sub>	UO <sub>2</sub>	<i>UO<sub>2</sub></i>
Pellet density (gm/cc) (% of theoretical)	<i>10.522</i> (96%)	<i>10.522</i> (96%)	<i>10.522</i> (96%)	<i>10.522</i> (96%)	<i>10.522</i> (96%)
Enrichment (wt.% <sup>235</sup> U)	3.4	<i>3.4</i>	3.4	3.4	<i>3.4</i>
Burnup (MWD/MTU)	40,000	<i>40,000</i>	40,000	40,000	<i>40,000</i>
Cooling time (years)	5	<i>5</i>	5	5	<i>5</i>
<i>Power/assembly (MW)</i>	<i>15.0</i>	<i>15.0</i>	<i>18.6</i>	<i>20.4</i>	<i>20.4</i>
Specific power (MW/MTU)	<i>36.409</i>	<i>41.097</i>	<i>39.356</i>	<i>43.031</i>	<i>47.137</i>
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	<i>467.319</i>	<i>414.014</i>	<i>536.086</i>	<i>537.752</i>	<i>490.901</i>
Weight of U (kg) <sup>†</sup>	<i>411.988</i>	<i>364.994</i>	<i>472.613</i>	<i>474.082</i>	<i>432.778</i>
No. of Guide Tubes	17	<i>17</i>	21	25	<i>25</i>
Guide Tube O.D. (in.)	0.539	<i>0.539</i>	0.546	0.474	<i>0.474</i>
Guide Tube Thickness (in.)	0.0170	<i>0.0170</i>	0.0170	0.0160	<i>0.0160</i>

<sup>†</sup> Derived from parameters in this table.

Table 5.2.25 (page 2 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
<i>Fuel assembly array class</i>	<i>14x14C</i>	<i>16x16A</i>	<i>15x15DEF H</i>	<i>17x17C</i>
Active fuel length (in.)	144	150	144	144
No. of fuel rods	176	236	208	264
Rod pitch (in.)	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	0.3805	0.3255	0.3742	0.3252
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (95% of theoretical)	<i>10.522</i> <i>(96%)</i>	<i>10.522</i> <i>(96%)</i>	10.412 (95%)	<i>10.522</i> <i>(96%)</i>
Enrichment (wt.% <sup>235</sup> U)	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
<i>Power/assembly (MW)</i>	<i>13.7</i>	<i>17.5</i>	<i>19.819</i>	<i>20.4</i>
Specific power (MW/MTU)	<i>31.275</i>	<i>39.083</i>	40	<i>42.503</i>
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	<i>496.887</i>	<i>507.9</i>	562.029	<i>544.428</i>
Weight of U (kg) <sup>†</sup>	<i>438.055</i>	<i>447.764</i>	495.485	<i>479.968</i>
No. of Guide Tubes	5	5	17	25
Guide Tube O.D. (in.)	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0400	0.0400	0.0160	0.0175

<sup>†</sup> Derived from parameters in this table.

Table 5.2.26 (page 1 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	7×7	8×8	8x8	9x9	9x9
<i>Fuel assembly array class</i>	<i>7x7B</i>	<i>8x8B</i>	<i>8x8CDE</i>	<i>9x9A</i>	<i>9x9B</i>
Active fuel length (in.)	144	144	150	144	150
No. of fuel rods	49	64	62	74	72
Rod pitch (in.)	0.738	0.642	0.64	0.566	0.572
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.484	0.493	0.44	0.433
Cladding thickness (in.)	0.0355	0.02725	0.034	0.028	0.026
Pellet diameter (in.)	0.488	0.4195	0.416	0.376	0.374
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (% of theoretical)	10.412 (95%)	10.412 (95%)	10.412 (95%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
<i>Power/assembly (MW)</i>	<i>5.96</i>	<i>5.75</i>	<i>5.75</i>	<i>5.75</i>	<i>5.75</i>
Specific power (MW/MTU)	30	30	30.24	31.97	31.88
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	225.177	217.336	215.673	204.006	204.569
Weight of U (kg) <sup>†</sup>	198.516	191.603	190.137	179.852	180.348
No. of Water Rods	0	0	2	2	1
Water Rod O.D. (in.)	n/a	n/a	0.493	0.98	1.516
Water Rod Thickness (in.)	n/a	n/a	0.034	0.03	0.0285

<sup>†</sup> Derived from parameters in this table.

Table 5.2.26 (page 1 of 2)

## DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	9x9	9x9	9x9	10x10	10x10
Fuel assembly array class	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Active fuel length (in.)	150	144	150	144	150
No. of fuel rods	80	76	72	92	96
Rod pitch (in.)	0.572	0.572	0.572	0.510	0.488
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.423	0.443	0.424	0.404	0.378
Cladding thickness (in.)	0.0295	0.0285	0.03	0.0260	0.0243
Pellet diameter (in.)	0.3565	0.3745	0.3565	0.345	0.3224
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc) (% of theoretical)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% <sup>235</sup> U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	5.75	5.75	5.75	5.75	5.75
Specific power (MW/MTU)	31.58	31.38	35.09	30.54	32.18
Weight of UO <sub>2</sub> (kg) <sup>†</sup>	206.525	207.851	185.873	213.531	202.687
Weight of U (kg) <sup>†</sup>	182.073	183.242	163.865	188.249	178.689
No. of Water Rods	1	5	1	2	1
Water Rod O.D. (in.)	0.512	0.546	1.668	0.980	Note 1
Water Rod Thickness (in.)	0.02	0.0120	0.032	0.0300	Note 1

Note 1: 10x10C has a diamond shaped water rod with 4 additional segments dividing the fuel rods into four quadrants.

<sup>†</sup> Derived from parameters in this table.

Table 5.2.27

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD PWR FUEL  
3.4 wt.% <sup>235</sup>U - 40,000 MWD/MTU - 5 years cooling

Assembly	WE 14×14	WE 14×14	WE 15×15	WE 17×17	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Array class	14x14A	14x14B	15x15 ABC	17x17A	17x17B	14x14C	16x16A	15x15 DEFH	17x17C
Neutrons/sec	1.76E+8 1.78E+8	2.32E+8 2.35E+8	2.70E+8 2.73E+8	2.18E+8	2.68E+8	2.32E+8	2.38E+8	2.94E+8	2.68E+8
Photons/sec (0.45-3.0 MeV)	2.88E+15 2.93E+15	3.28E+15 3.32E+15	3.80E+15 3.86E+15	3.49E+15	3.85E+15	3.37E+15	3.57E+15	4.01E+15	3.89E+15
Thermal power (watts)	809.5 820.7	923.5 933.7	1073 1086	985.6	1090	946.6	1005	1137	1098

## Note:

The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.28

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD BWR FUEL  
 3.0 wt.% <sup>235</sup>U - 40,000 MWD/MTU - 5 years cooling

Assembly	7x7	8x8	8x8	9x9	9x9	9x9	9x9	9x9	10x10	10x10
Array Class	7x7B	8x8B	8x8CDE	9x9A	9x9B	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Neutrons/sec	1.33E+8	1.22E+8	1.22E+8	1.13E+8	1.06E+8	1.09E+8	1.24E+8	9.15E+7	1.24E+8	1.07E+8
Photons/sec (0.45-3.0 MeV)	1.55E+15	1.49E+15	1.48E+15	1.41E+15	1.40E+15	1.42E+15	1.45E+15	1.28E+15	1.48E+15	1.40E+15
Thermal power (watts)	435.5	417.3	414.2	394.2	389.8	395	405.8	356.9	413.5	389.2



Table 5.2.29

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL  
AND VALUES REPORTED IN THE  
DOE CHARACTERISTICS DATABASE<sup>†</sup> FOR  
30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Design Source Term Calculations Basis Fuel (watts/assembly)
<b>PWR Fuel</b>		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	<del>827.5</del> 802.7
CE 16x16	653.7	<del>827.5</del> 734.3
CE 14x14	601.3	<del>827.5</del> 694.9
WE 17x17	742.5	<del>827.5</del> 795.4
WE 15x15	762.2	<del>827.5</del> 796.2
WE 14x14	649.6	<del>827.5</del> 682.9
<b>BWR Fuel</b>		
7x7	310.9	315.7
8x8	296.6	<del>315.7</del> 302.8
9x9	275.0	<del>315.7</del> 286.8

Notes:

1. The ~~PWR and BWR design basis fuels are the B&W 15x15 and the GE 7x7, respectively~~ decay heat from the source term calculations is the maximum value calculated for that fuel assembly class.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.
4. *The enrichments used for the column labeled "Decay Heat from Source Term Calculations" were consistent with Table 5.2.24.*

<sup>†</sup> Reference [5.2.7].

Table 5.2.30

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY  
AND THIMBLE PLUG DEVICE

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

Table 5.2.31

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD  
ASSEMBLIES AND THIMBLE PLUG DEVICES

<b>Region</b>	<b>BPRA</b>	<b>TPD</b>
Upper End Fitting (curies Co-60)	30.432.7	25.21
Gas Plenum Spacer (curies Co-60)	4.65.0	9.04
Gas Plenum Springs (curies Co-60)	8.28.9	15.75
In-core (curies Co-60)	787.8848.4	N/A

Table 5.2.32

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY  
CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Inconel)	Mass of absorber (kg AgInCd)
Start (in)	Finish (in)	Length (in)			
<b>Configuration 1 - 10% Inserted</b>					
0.0	15.0	15.0	1.0	1.32	7.27
15.0	18.8125	3.8125	0.2	0.34	1.85
18.8125	28.25	9.4375	0.1	0.83	4.57
<b>Configuration 2 - Fully Removed</b>					
0.0	3.8125	3.8125	0.2	0.34	1.85
3.8125	13.25	9.4375	0.1	0.83	4.57

Table 5.2.33

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD  
CONFIGURATION S FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Steel)	Mass of absorber (kg Inconel)
Start (in)	Finish (in)	Length (in)			
<b>Configuration 1 - 10% Inserted</b>					
0.0	15.0	15.0	1.0	1.26	5.93
15.0	18.8125	3.8125	0.2	0.32	1.51
18.8125	28.25	9.4375	0.1	0.79	3.73
<b>Configuration 2 - Fully Removed</b>					
0.0	3.8125	3.8125	0.2	0.32	1.51
3.8125	13.25	9.4375	0.1	0.79	3.73
<b>Configuration 3 - Fully Inserted</b>					
0.0	63.0	63.0	1.0	5.29	24.89
63.0	66.8125	3.8125	0.2	0.32	1.51
66.8125	76.25	9.4375	0.1	0.79	3.73

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD  
ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Photons/sec from AgInCd			Curies Co-60 from Inconel
Start (in)	Finish (in)	Length (in)	0.3-0.45 MeV	0.45-0.7 MeV	0.7-1.0 MeV	
<b>Configuration 1 - 10% Inserted - 80.8 watts decay heat</b>						
0.0	15.0	15.0	1.91e+14	1.78e+14	1.42e+14	1111.38
15.0	18.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
18.8125	28.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92
<b>Configuration 2 - Fully Removed - 8.25 watts decay heat</b>						
0.0	3.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
3.8125	13.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER  
SHAPING ROD CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Curies of Co-60
Start (in)	Finish (in)	Length (in)	
<b>Configuration 1 - 10% Inserted - 46.2 watts decay heat</b>			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78
<b>Configuration 2 - Fully Removed - 4.72 watts decay heat</b>			
0.0	3.8125	3.8125	136.36
3.8125	13.25	9.4375	168.78
<b>Configuration 3 - Fully Inserted - 178.9 watts decay heat</b>			
0.0	63.0	63.0	11266.80
63.0	66.8125	3.8125	136.36
66.8125	76.25	9.4375	168.78

Table 5.2.36

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

	<b>BWR</b>
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO <sub>2</sub> fuel rods	55
No. of UO <sub>2</sub> /ThO <sub>2</sub> 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 <sub>2</sub> and 1.8% UO <sub>2</sub> for UO <sub>2</sub> /ThO <sub>2</sub> rods
Pellet density (gm/cc)	10.412
Enrichment (w/o <sup>235</sup> U)	93.5 in UO <sub>2</sub> for UO <sub>2</sub> /ThO <sub>2</sub> rods and 1.8 for UO <sub>2</sub> rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of ThO <sub>2</sub> and UO <sub>2</sub> (kg) <sup>†</sup>	121.46
Weight of U (kg) <sup>†</sup>	92.29
Weight of Th (kg) <sup>†</sup>	14.74

<sup>†</sup> Derived from parameters in this table.



Table 5.2.37

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

Lower Energy	Upper Energy	16,000 MWD/MTIHM 18-Year Cooling	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
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	1.09e+12

Table 5.2.38

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

<b>Lower Energy (MeV)</b>	<b>Upper Energy (MeV)</b>	<b>16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)</b>
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

### 5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM 100 System, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Section 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

#### 5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM 100 System, including the HI-TRAC transfer casks. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 through 5.3.6 show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. Figures 5.3.1 through 5.3.3 were created with the MCNP two-dimensional plotter and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figure 5.3.7 shows a cross sectional view of the 100-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. These dimensions are the same for the HI-STORM 100S overpack. Figures 5.3.11 and 5.3.18 are axial representations of the HI-STORM 100 and HI-STORM 100S overpacks, respectively, with the various as-modeled dimensions indicated.

Figures 5.3.12 and 5.3.13 show axial cross-sectional views of the 100- and 125-ton HI-TRAC transfer casks, respectively, with the as-modeled dimensions and materials specified. Figures 5.3.14, 5.3.15, and 5.3.20 show fully labeled radial cross-sectional views of the HI-TRAC 100, 125, and 125D transfer casks, respectively. Finally, Figures 5.3.16 and 5.3.17 show fully labeled

diagrams of the transfer lids for the HI-TRAC 100 and 125 transfer casks. Since lead plate may be used instead of poured lead in the pool and transfer lids, there exists the possibility of a gap between the lead plate and the surrounding steel walls. This gap was accounted for in the analysis as depicted on Figures 5.3.16 and 5.3.17. The gap was not modeled in the pool lid since the gap will only exist on the outer edges of the pool lid and the highest dose rate is in the center. (All results presented in this chapter were calculated with the gap with the exception of the results presented in Figures 5.1.6, 5.1.7, and 5.1.11 which did not include the gap.) The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid in this chapter are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. Consistent with the analysis of the transfer lid in which only the portion of the lid directly below the MPC was modeled, the structure of the mating device which surrounds the pool lid was not modeled.

Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. The HI-TRAC 125D has steel ribs in the lower water jacket while the HI-TRAC 125 does not. These additional ribs in the lower water jacket reduce the dose rate in the vicinity of the pool lid for the HI-TRAC 125D compared to the HI-TRAC 125. Since the dose rates at the midplane of the HI-TRAC 125D are higher than the HI-TRAC 125, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.

To reduce the gamma dose around the inlet and outlet vents, stainless steel cross plates, designated gamma shield cross plates<sup>†</sup> (see Figures 5.3.11 and 5.3.18), have been installed inside all vents. The steel in these plates effectively attenuates the fuel and <sup>60</sup>Co gammas that dominated the dose at these locations prior to their installation. Figure 5.3.19 shows two designs for the gamma shield cross plates to be used in the inlet and outlet vents. The designs in the top portion of the figure are mandatory for use in the HI-STORM 100 and 100S overpacks during normal storage operations and were assumed to be in place in the shielding analysis. The designs in the bottom portion of the figure may be used instead of the mandatory designs in the HI-STORM 100S overpack to further reduce the radiation dose rates at the vents. These optional gamma shield cross plates could further reduce the dose rate at the vent openings by as much as a factor of two.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 15 times the pitch and 7 times the pitch, respectively. Homogenization resulted in a noticeable decrease in run time.

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<sup>†</sup> This design embodiment, formally referred to as “Duct Photon Attenuator,” has been disclosed as an invention by Holtec International for consideration by the US Patent Office for issuance of a patent under U.S. law.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The basket material in the top and bottom 0.9 inches where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9-inch area at the top and bottom of the MPC basket.
2. The upper and lower fuel spacers are not modeled, as the fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The fuel spacer length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis, but the fuel spacer materials are not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
3. For the MPC-32, MPC-24, and MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel that would provide a small increase in shielding. The optional aluminum heat conduction elements are also conservatively not modeled.
4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel that would provide a small amount of additional shielding.
5. In the modeling of the BWR fuel assemblies, the zircaloy flow channels were not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.
6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches compared to 6.25 inches or 7.5 inches.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STORM 100 and HI-TRAC. Therefore, the models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

### MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in both the HI-TRAC and the HI-STORM outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-TRAC or the HI-STORM.
2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Since the new MPC-24 configuration and the configuration of the MPC-24E are almost identical, the analysis of the earlier MPC-24 configuration is valid for the MPC-24E as well. Figure 5.3.21 shows the superseded and current configuration for the MPC-24 for comparison.
3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches to 0.0235 inches. However, the model still uses 0.06 inches. This discrepancy is compensated for by the use of 9/32 inch cell walls and 5 inch boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration in the 100-ton HI-TRAC for comparison to the superseded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface. These results demonstrate that using the superseded MPC-24 design is conservative.

### HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the HI-TRAC 125 was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces effective shielding thickness. The HI-TRAC 125D does not use pocket trunnions.
2. The lifting blocks in the top lid of the 125-ton HI-TRACs were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.
3. The door side plates that are in the middle of the transfer lid of the HI-TRAC 125 are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.
4. The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRACs was modeled as 4 inches larger than it is due to a design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.

### HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.
3. In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.
4. The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.
5. The pedestal baseplate, which is steel with holes for pouring concrete, in the HI-STORM overpacks was modeled as concrete rather than steel. This is acceptable because this piece of steel is positioned at the bottom of the pedestal below 5 inches of steel and a minimum of 11.5 inches of concrete and therefore will have no impact on the dose rates at the bottom vent.
6. Minor penetrations in the body of the overpack (e.g. holes for grounding straps) are not modeled as these are small localized effects which will not affect the off-site dose rates.
7. In June 2001, the inner shield shell of the HI-STORM 100 overpack was removed and the concrete density in the body of the overpack (not the pedestal of lid) was increased to compensate. Appendix 5.E presents a comparison of the dose rates calculated for a HI-STORM 100 overpack with and without the inner shield shell. The MPC-24 was used in this comparison. The results indicate that there is very little difference in the calculated dose rates when the inner shield shell is removed and the concrete density is increased. Therefore, all HI-STORM 100 analysis presented in the main portion of this chapter includes the inner shield shell.
8. The drawings in Section 1.5 indicate that the HI-STORM 100S has a variable height. This is achieved by adjusting the height of the body of the overpack. The pedestal height is not adjusted. Conservatively, all calculations in this chapter used the shorter height for the HI-STORM 100S.

9. In February 2002, the top plate on the HI-STORM 100 overpack was modified to be two pieces in a shear ring arrangement. The total thickness of the top plate was not changed. However, there is approximately a 0.5 inch gap between the two pieces of the top plate. This gap was not modeled in MCNP since it will result in a small increase in the dose rate on the overpack lid in an area where the dose rate is greatly reduced compared to other locations on the lid.

#### 5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, basket, inlet vents, and outlet vents are shown in these figures.

#### 5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

The MCNP model of the HI-TRAC transfer cask describes the lifting trunnions, pocket trunnions, and the opening in the HI-TRAC top lid. The fins through the HI-TRAC water jacket are also modeled. Streaming considerations through these trunnions and fins are discussed in Section 5.4.1.

The design of the HI-STORM 100 System, as described in the drawings in Chapter 1, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the HI-TRAC transfer cask.
- The pocket trunnions of the HI-TRAC are modeled as solid blocks of steel. No credit is taken for any part of the pocket trunnion that extends beyond the water jacket.
- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.
- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. The holes in the vent and drain port cover plates are filled with a set screw and plug weld. The steel lost in the MPC lid at the port location is replaced with a block of steel



approximately 6 inches thick located directly below the port opening and attached to the underside of the lid. This design feature is shown on the drawings in Chapter 1. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid plate.

### 5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM 100 System and HI-TRAC shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

Since the HI-STORM 100S and the newer configuration of the HI-STORM 100 do not have the inner shield shell present, the minimum density of the concrete in the body (not the lid or pedestal) of the overpack has been increased slightly to compensate for the change in shielding relative to the HI-STORM 100 overpack with the inner shield shell. Table 5.3.2 shows the concrete composition and densities that were used for the HI-STORM 100 and HI-STORM 100S overpacks. Since the density of concrete is increased by altering the aggregate that is used, the composition of the slightly denser concrete was calculated by keeping the same mass of water as the 2.35 gm/cc composition and increasing all other components by the same ratio.

*The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorbing materials: Boral or Metamic. Both materials are made of aluminum and  $B_4C$  powder. The Boral contains an aluminum and  $B_4C$  powder mixture sandwiched between two aluminum plates while the Metamic is a single plate. The thickness and minimum  $^{10}B$  areal density are the same for Boral and Metamic. Therefore, the mass of Aluminum and  $B_4C$  are essentially equivalent and there is no distinction between the two materials from a shielding perspective. As a result, Table 5.3.2 identifies the composition for Boral and no explicit calculations were performed with Metamic.*

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

*Table 4.4.36 indicates that there are localized areas in the concrete in the lid of the overpack which approach 339 °F. A bounding increase in temperature from 300 °F to 365 °F results in an approximate 0.424% overall density reduction due to the loss of chemically unbound water. This density reduction results in a reduction in the mass fraction of hydrogen from 0.6% to 0.555% in the area affected by the temperature excursion. This is a localized effect with the maximum loss occurring at the bottom center of the lid where the temperature is the hottest and reduced loss occurring as the temperature decreases to 300 °F.*

*Based on these considerations, the presence of localized temperatures in excess of 300°F in the lid concrete has a negligible effect on the shielding effectiveness of the HI-STORM 100 overpack lid.*

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES<sup>†</sup>

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
<b>PWR</b>					
Lower End Fitting	0.0	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
<b>BWR</b>					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

<sup>†</sup> All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

## COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Uranium Oxide	10.412	<sup>235</sup> U	2.9971(BWR) 3.2615(PWR)
		<sup>238</sup> U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral <sup>†</sup>	2.644	<sup>10</sup> B	4.4226 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM)4.367 (MPC-24 in HI-TRAC)
		<sup>11</sup> B	20.1474 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

<sup>†</sup> All B-10 loadings in the Boral compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		<sup>10</sup> B	0.14087
		<sup>11</sup> B	0.64174
BWR Fuel Region Mixture	4.29251	<sup>235</sup> U	2.4966
		<sup>238</sup> U	70.9315
		O	9.8709
		Zr	16.4046
		N	8.35E-05
		Cr	0.0167
		Fe	0.0209
		Sn	0.2505
PWR Fuel Region Mixture	3.869939	<sup>235</sup> U	2.7652
		<sup>238</sup> U	71.9715
		O	10.0469
		Zr	14.9015
		Cr	0.0198
		Fe	0.0365
		Sn	0.2587

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

<b>Component</b>	<b>Density (g/cm<sup>3</sup>)</b>	<b>Elements</b>	<b>Mass Fraction (%)</b>
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.4862	SS304	100
Gas Plenum Springs (BWR)	0.2653	SS304	100
Expansion Springs (BWR)	0.6775	SS304	100
Upper End Fitting (BWR)	1.3692	SS304	100
Handle (BWR)	0.2572	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.9140 (water jacket)	H	11.2
	0.9619 (inside MPC)	O	88.8

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Concrete  Lid and pedestal of the HI-STORM 100 and 100S and the body of the 100 when the inner shield shell is present	2.35	H	0.6
		O	50.0
		Si	31.5
		Al	4.8
		Na	1.7
		Ca	8.3
		Fe	1.2
		K	1.9
Concrete  HI-STORM 100S body and HI-STORM 100 body when the inner shield shell is not present	2.48	H	0.569
		O	49.884
		Si	31.594
		Al	4.814
		Na	1.705
		Ca	8.325
		Fe	1.204
		K	1.905

Table 5.3.3

COMPOSITION OF THE FUEL PELLETS IN THE MIXED OXIDE FUEL ASSEMBLIES

Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Mixed Oxide Pellets	10.412	<sup>238</sup> U	85.498
		<sup>235</sup> U	0.612
		<sup>238</sup> Pu	0.421
		<sup>239</sup> Pu	1.455
		<sup>240</sup> Pu	0.034
		<sup>241</sup> Pu	0.123
		<sup>242</sup> Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	<sup>238</sup> U	86.175
		<sup>235</sup> U	1.975
		O	11.85



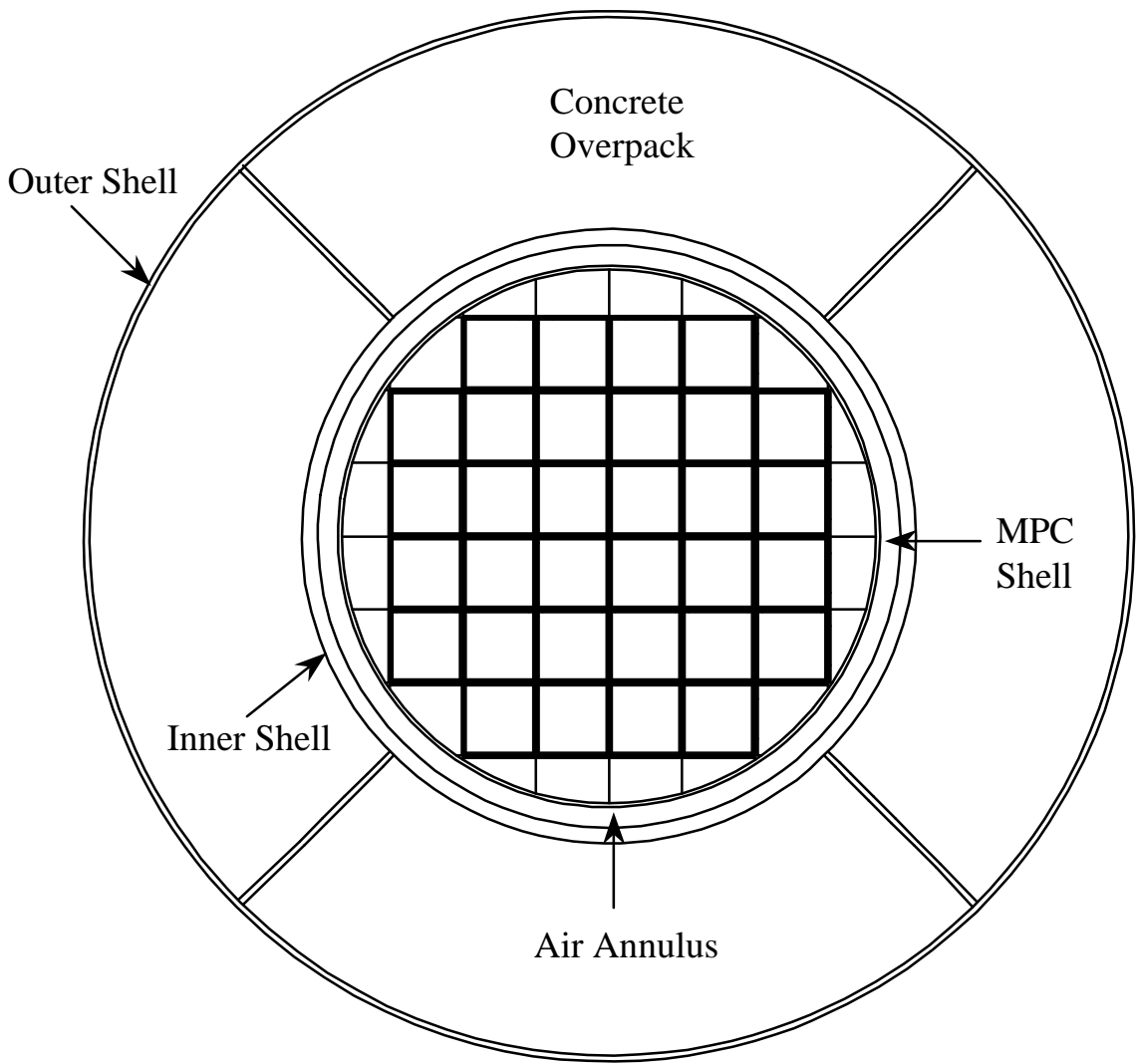


FIGURE 5.3.1; HI-STORM 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>

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<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

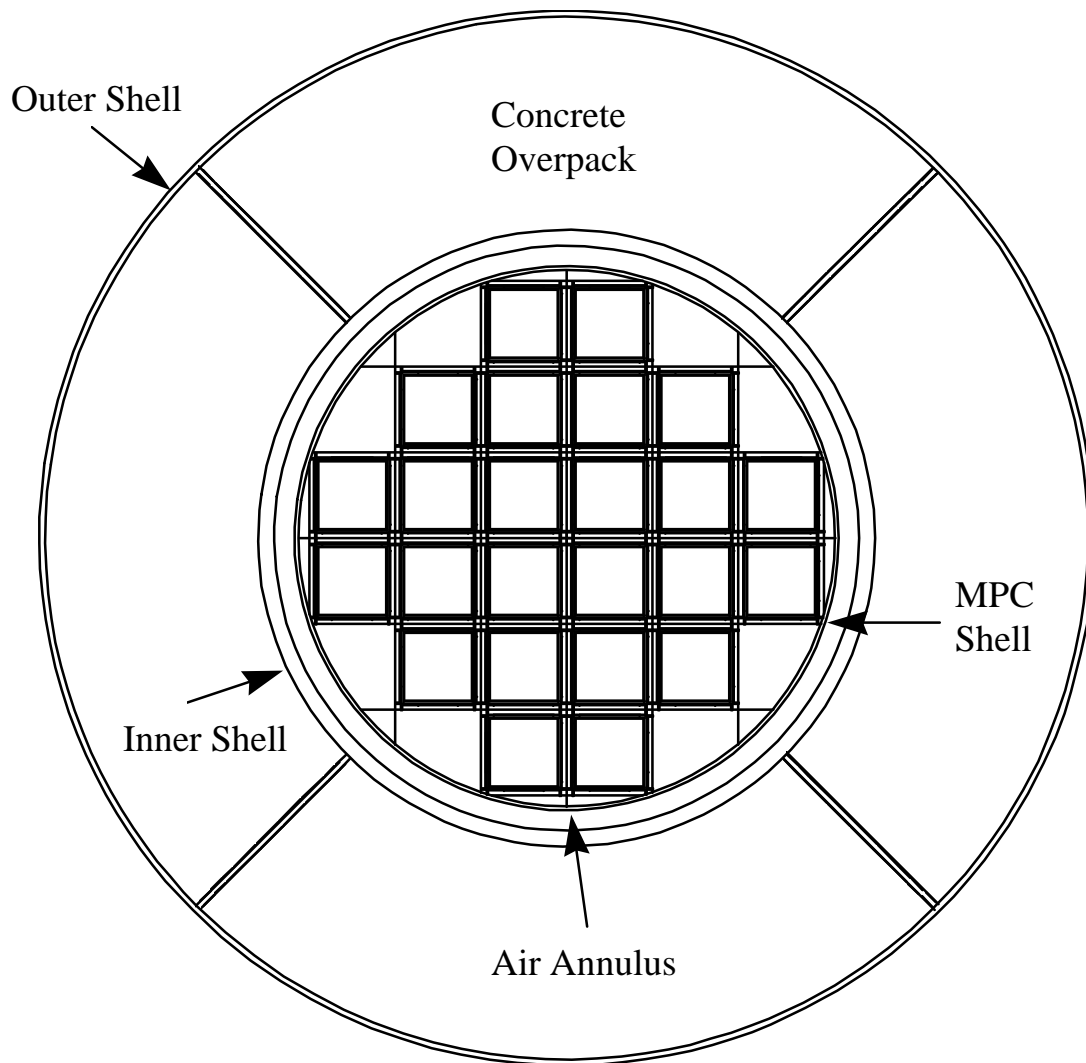


FIGURE 5.3.2; HI-STORM 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

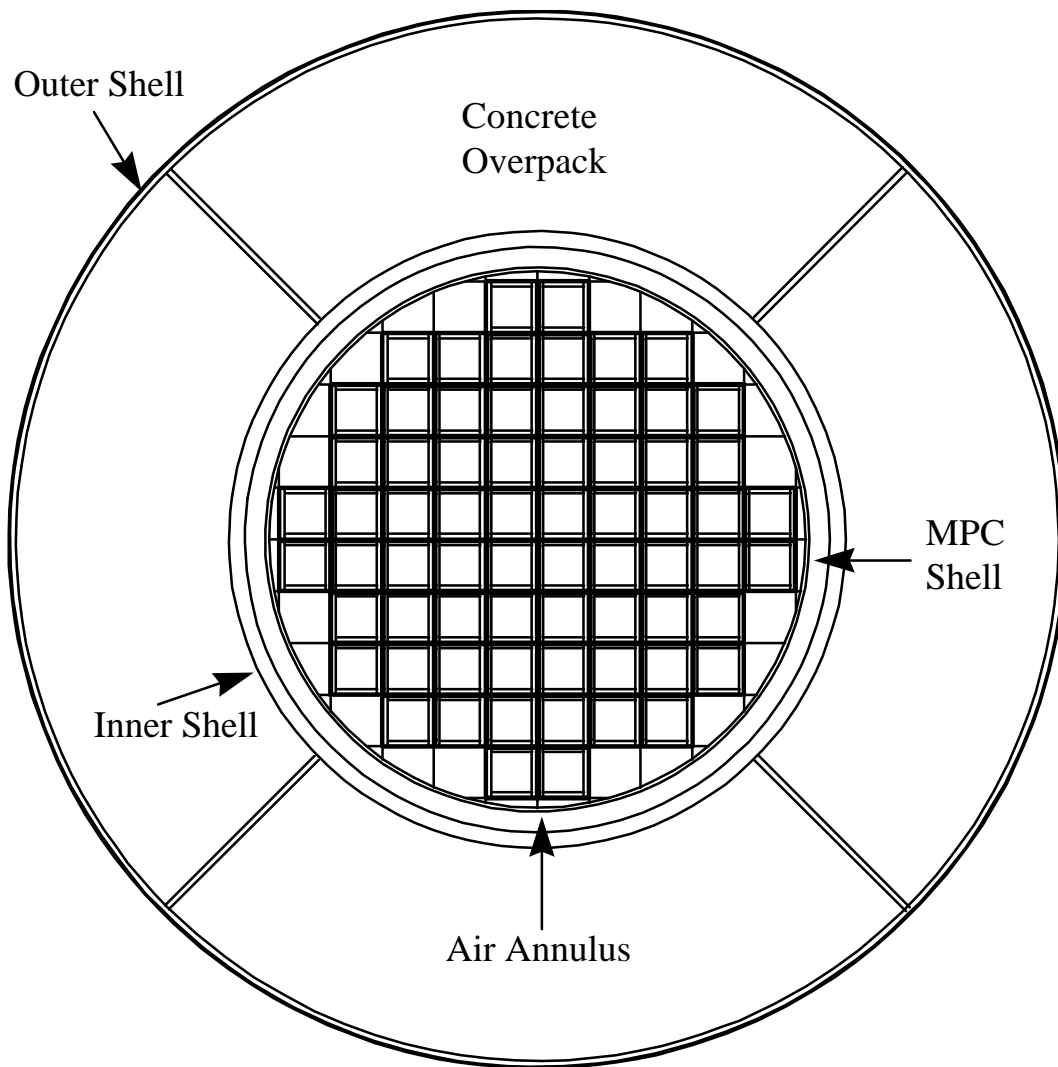


FIGURE 5.3.3; HI-STORM 100 OVERPACK WITH MPC-68 CROSS SECTIONAL VIEW AS MODELLED IN MCNP<sup>†</sup>

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

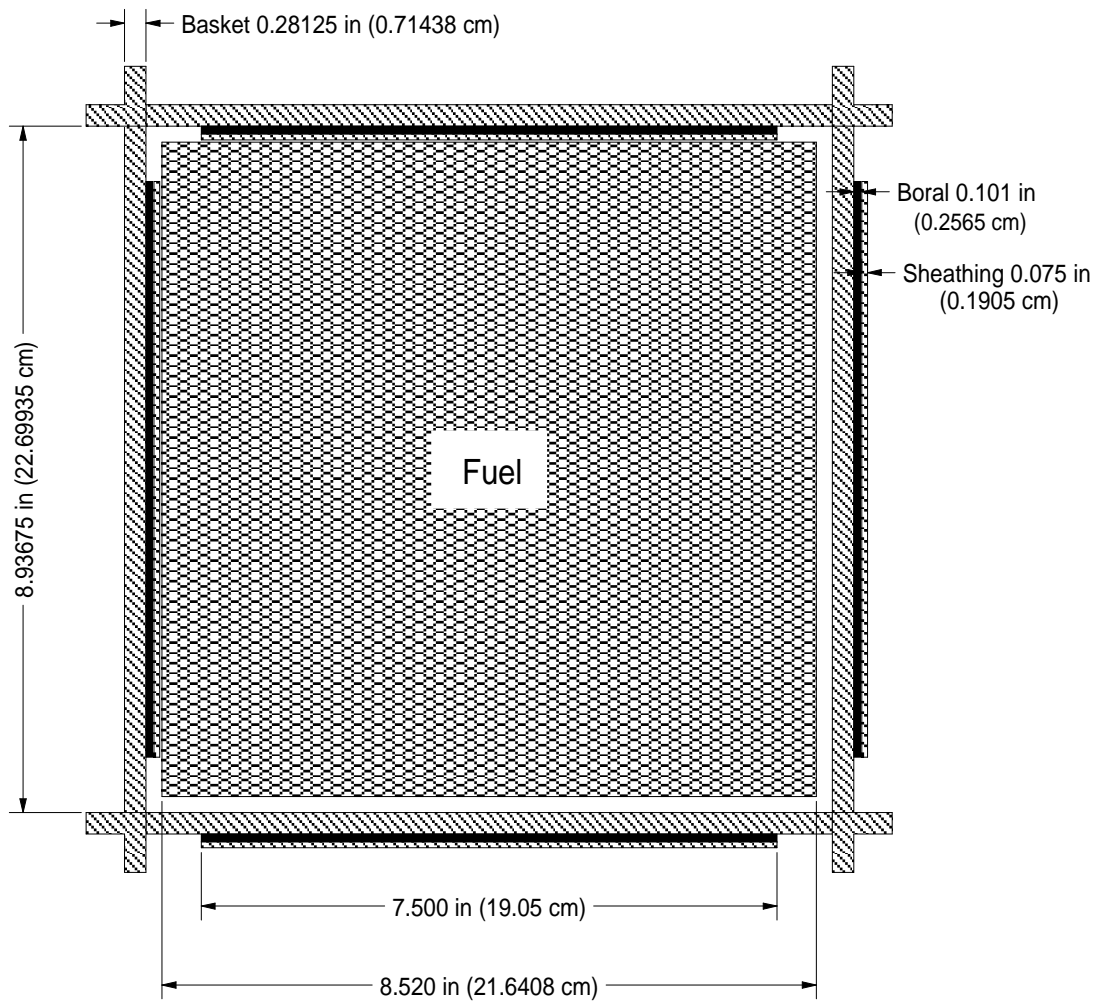


FIGURE 5.3.4; CROSS SECTIONAL VIEW OF AN MPC-32 BASKET CELL AS MODELED IN MCNP

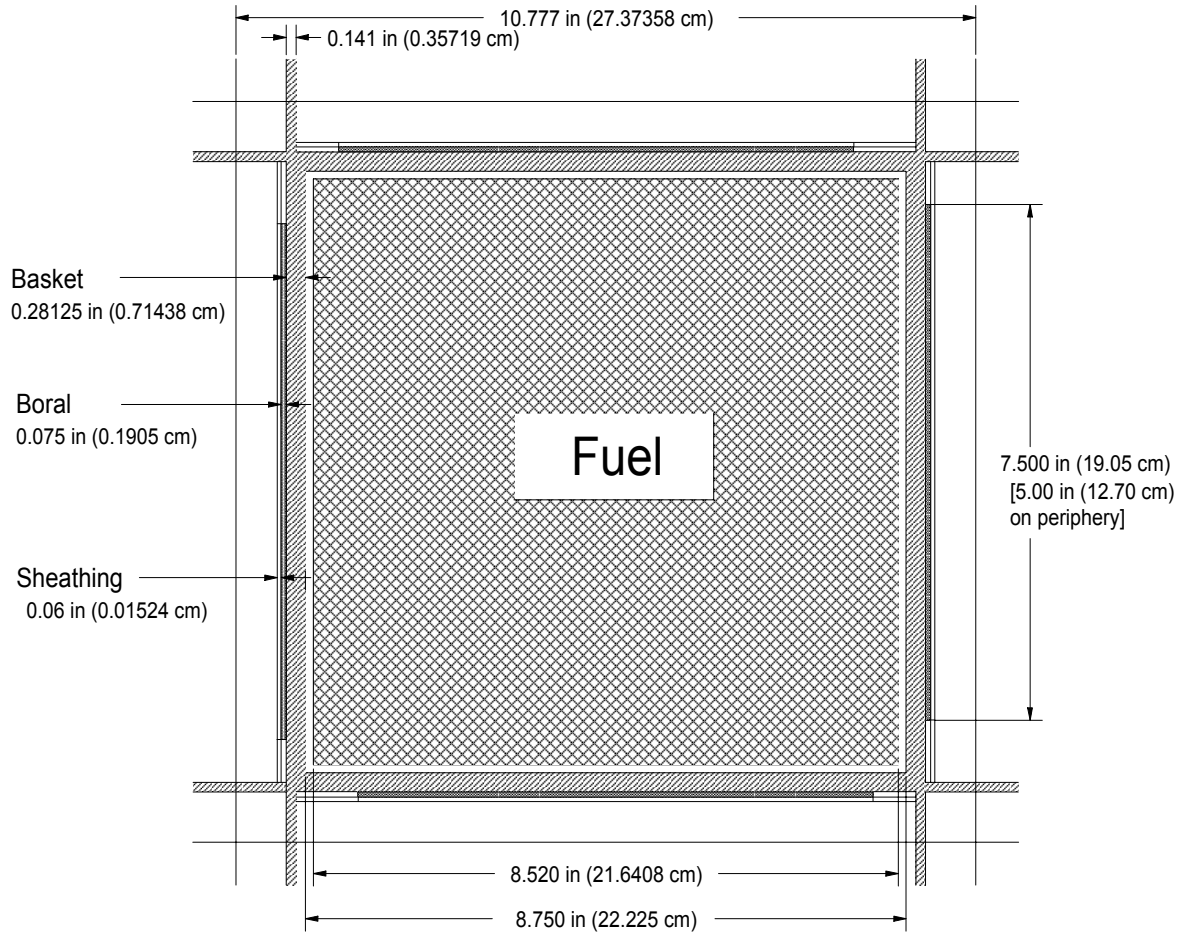


FIGURE 5.3.5; CROSS SECTIONAL VIEW OF AN MPC-24 BASKET CELL AS MODELED IN MCNP

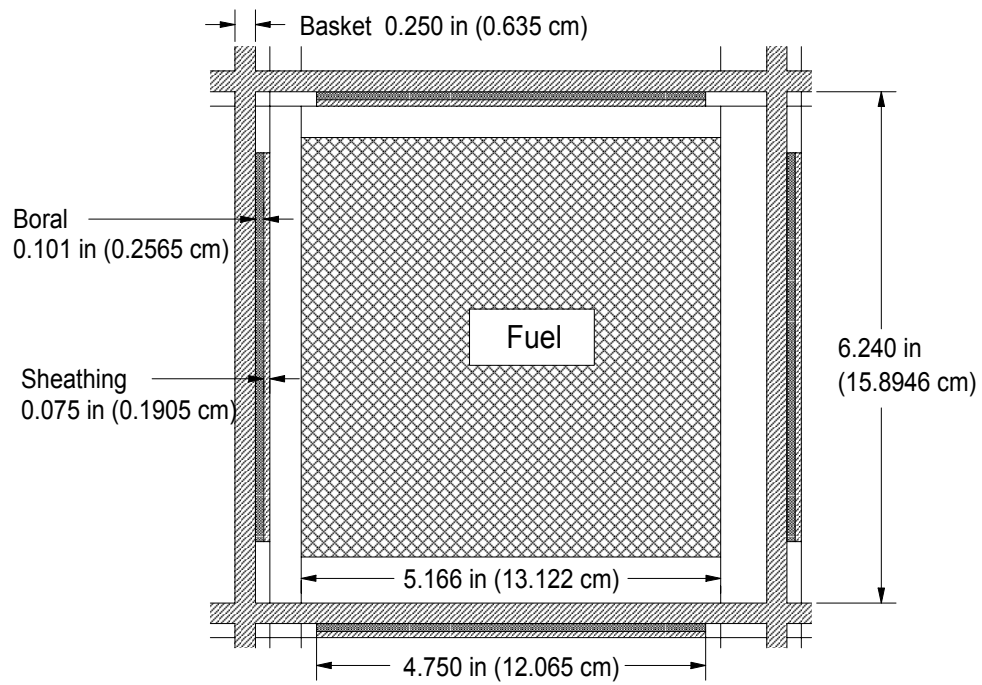


FIGURE 5.3.6; CROSS SECTIONAL VIEW OF AN MPC-68 BASKET CELL AS MODELED IN MCNP

# 100 TON HI-TRAC

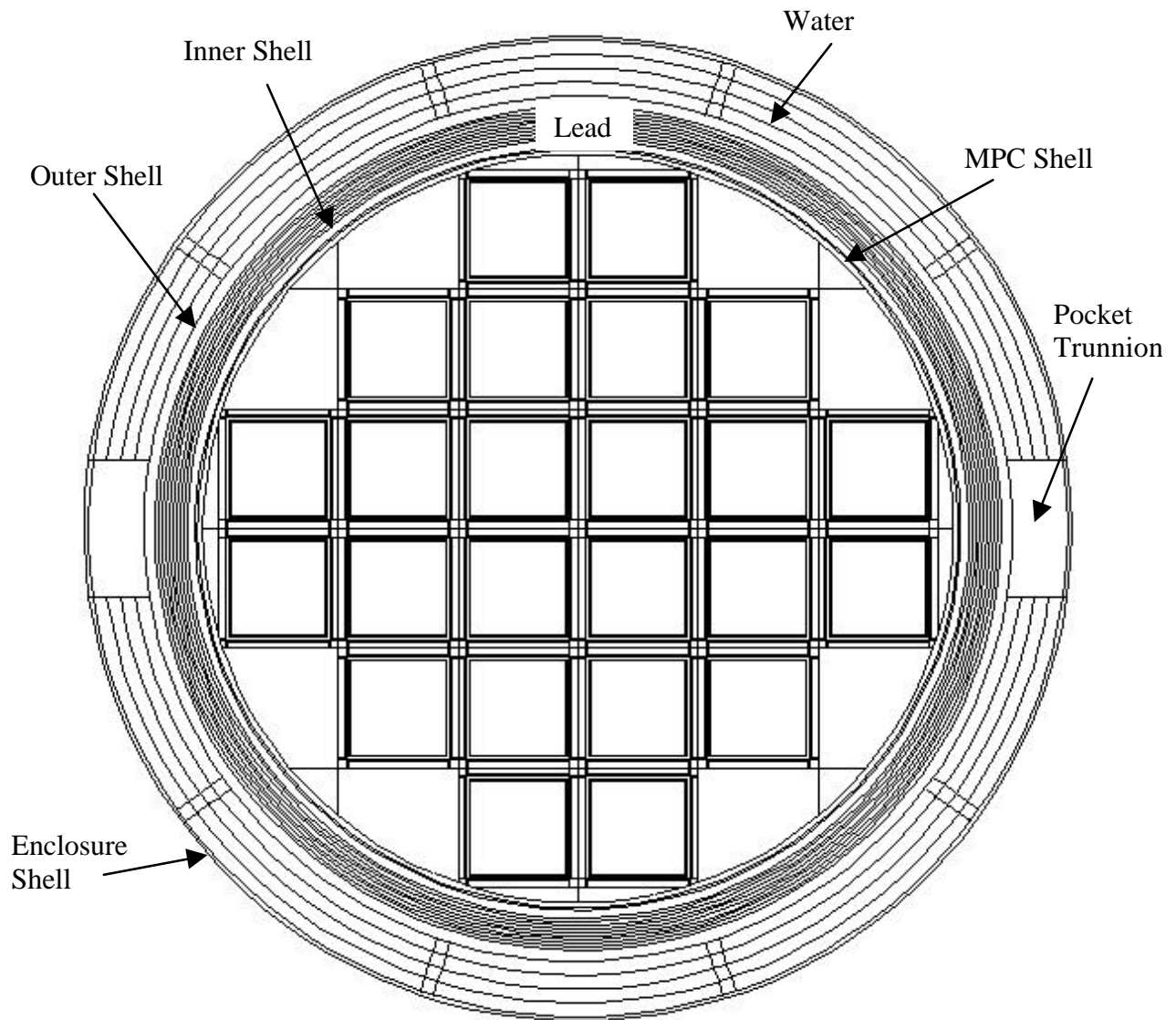


FIGURE 5.3.7; HI-TRAC OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELED IN MCNP<sup>†</sup>

<sup>†</sup> This figure is drawn to scale using the MCNP plotter.

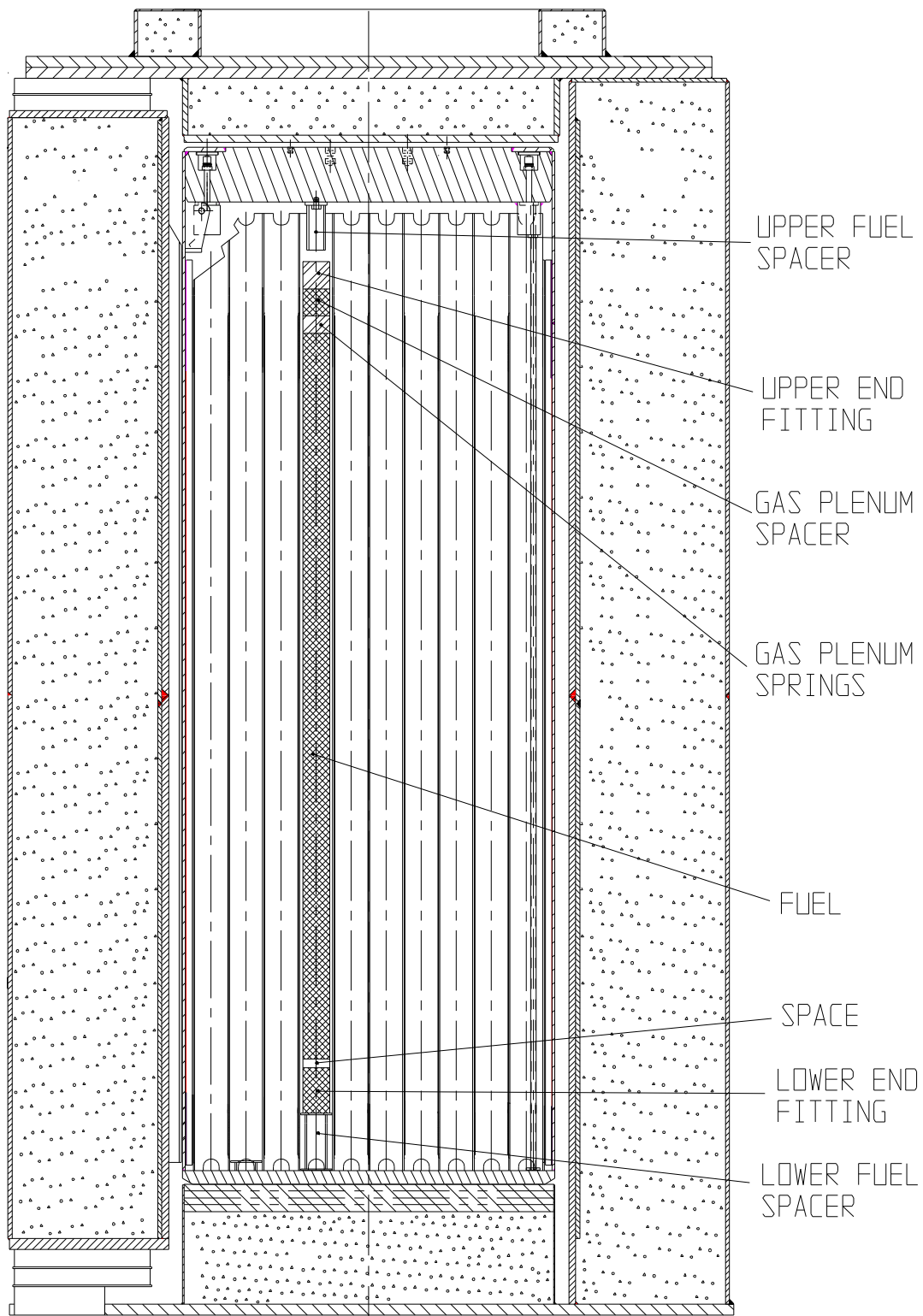


FIGURE 5.3.8; AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM OVERPACK



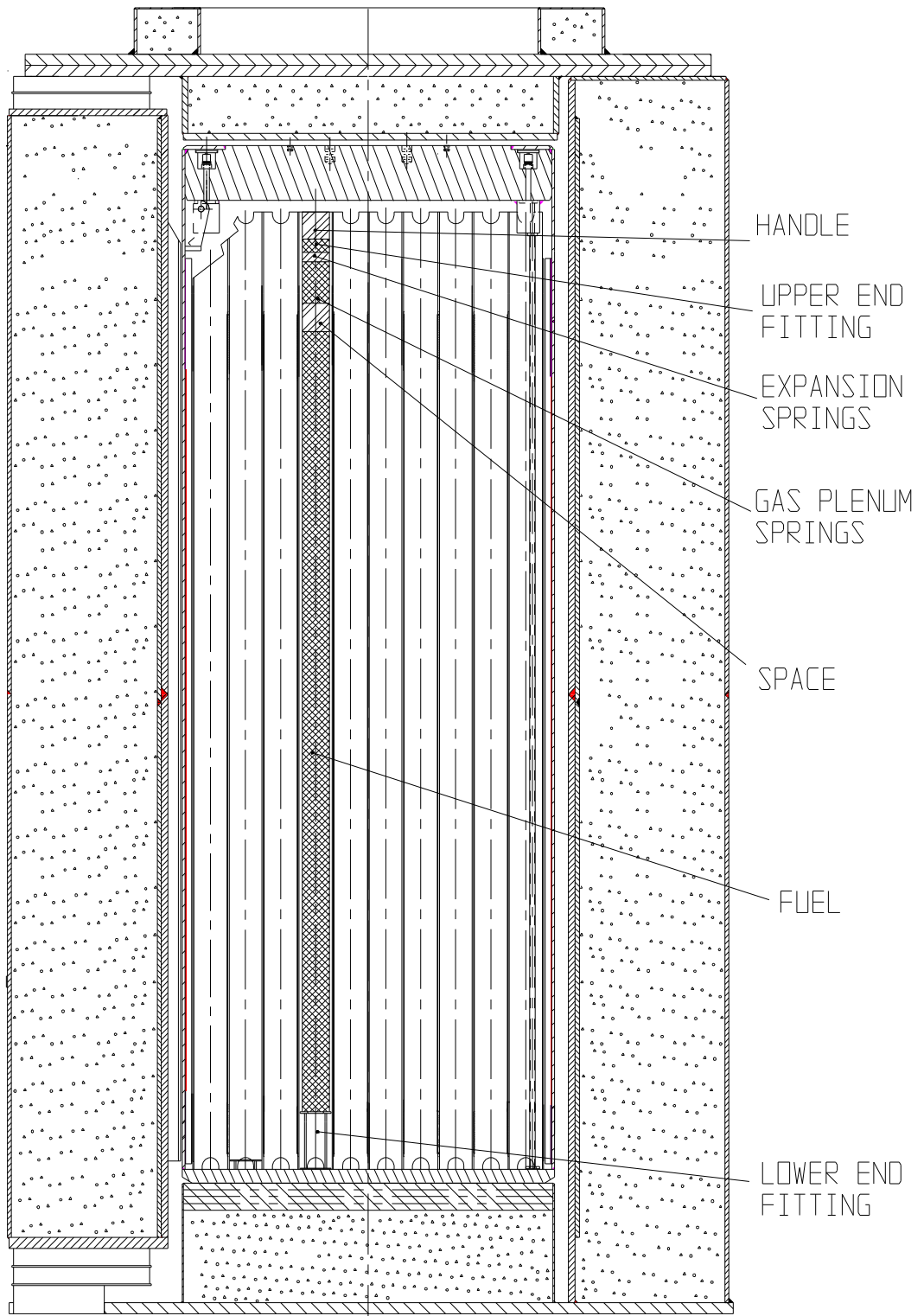


FIGURE 5.3.9; AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STORM OVERPACK

Figure withheld under 10 CFR 2.390

FIGURE 5.3.10; CROSS SECTION OF HI-STORM 100 OVERPACK

**Figure withheld under 10 CFR 2.390**

FIGURE 5.3.11; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

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Figure withheld under 10 CFR 2.390

FIGURE 5.3.12; 100-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS  
SECTIONAL ELEVATION VIEW (AS MODELED)

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Figure withheld under 10 CFR 2.390

FIGURE 5.3.13; 125-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS  
SECTIONAL ELEVATION VIEW (AS MODELED)

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**Figure withheld under 10 CFR 2.390**

**FIGURE 5.3.14; HI-TRAC 100 TRANSFER CASK CROSS SECTIONAL VIEW  
(AS MODELED)**

**Figure withheld under 10 CFR 2.390**

FIGURE 5.3.15; HI-TRAC 125 TRANSFER CASK CROSS SECTIONAL VIEW  
(AS MODELED)

Figure withheld under 10 CFR 2.390

FIGURE 5.3.16; 100-TON HI-TRAC TRANSFER LID (AS MODELED)

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Figure withheld under 10 CFR 2.390

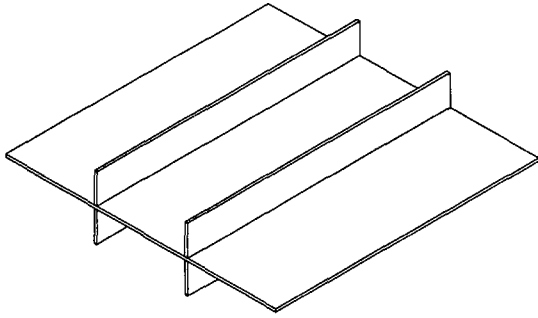
FIGURE 5.3.17; 125-TON HI-TRAC TRANSFER LID (AS MODELED)

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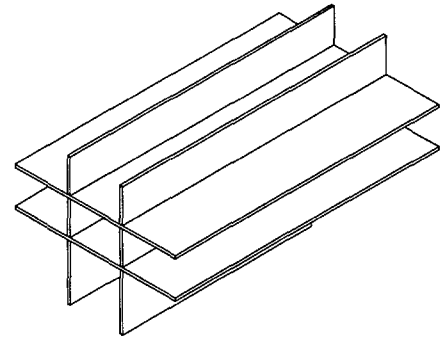
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Figure withheld under 10 CFR 2.390

FIGURE 5.3.18; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

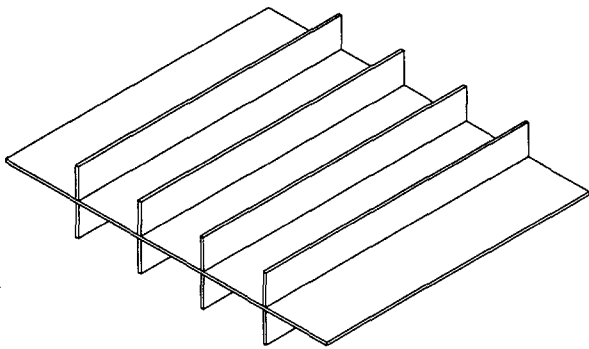


OUTLET VENT

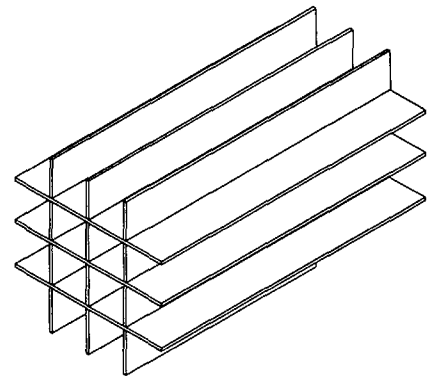


INLET VENT

MANDATORY GAMMA SHIELD CROSS PLATES FOR HI-STORM 100  
AND HI-STORM 100S



OUTLET VENT



INLET VENT

OPTIONAL GAMMA SHIELD CROSS PLATES FOR HI-STORM 100S  
THAT MAY BE USED INSTEAD OF THE MANDATORY DEVICES.

FIGURE 5.3.19: GAMMA SHIELD CROSS PLATE CONFIGURATION OF  
HI-STORM 100 AND HI-STORM 100S

Figure withheld under 10 CFR 2.390

FIGURE 5.3.20; HI-TRAC 125D TRANSFER CASK CROSS SECTIONAL VIEW  
(AS MODELED)

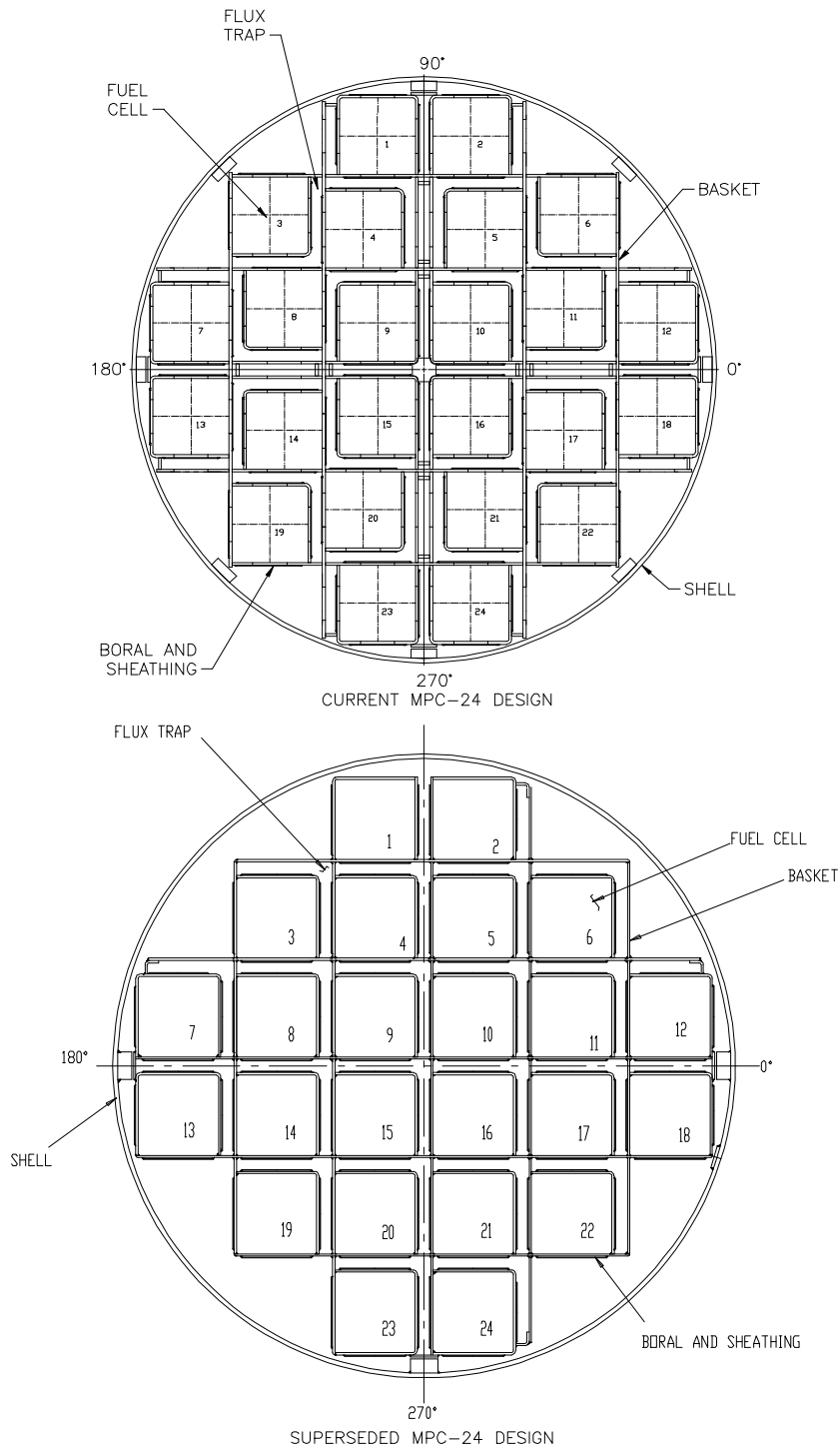


FIGURE 5.3.21; CROSS SECTIONAL VIEWS OF THE CURRENT MPC-24 DESIGN AND THE SUPERSEDED MPC-24 WHICH IS USED IN THE MCNP MODELS.

## 5.4 SHIELDING EVALUATION

The MCNP-4A code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and  $^{60}\text{Co}$ ). The axial distribution of the fuel source term is described in Table 2.1.11 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The  $^{60}\text{Co}$  source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.11 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ( $1.105^{4.2}/1.105$ ) and 76.8% ( $1.195^{4.2}/1.195$ ) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.C. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

*The dose rate at the various locations were calculated with MCNP using a two step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in the fuel and each axial location in the end fittings. The second and last step was to multiply the dose rate per starting particle for each group or starting location by the source strength (i.e. particles/sec) in that group or location and sum the resulting dose*

*rates for all groups in each dose location. The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location.*

The HI-STORM shielding analysis was performed for conservative burnup and cooling time combinations which bound the uniform and regionalized loading specifications for zircaloy clad fuel specified in ~~Appendix B to the CoC~~Section 2.1.9. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

Tables 5.1.1 through 5.1.3 provide the maximum dose rates adjacent to the HI-STORM overpack during normal conditions for each of the MPCs. Tables 5.1.4 through 5.1.6 provide the maximum dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Sections 5.1.1 and 5.1.2.

Tables 5.1.7 and 5.1.8 provide dose rates for the 100-ton and 125-ton HI-TRAC transfer casks, respectively, with the MPC-24 loaded with design basis fuel in the normal condition, in which the MPC is dry and the HI-TRAC water jacket is filled with water. Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with the water jacket filled with water (condition in which welding operations are performed). Dose locations 4 and 5, which are on the top and bottom of the HI-TRAC were not calculated at the one-meter distance for these configurations. For the conditions involving a fully flooded MPC, the internal water level was 10 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.7 indicates that the dose rates in the upper and lower portions of the HI-TRAC are reduced by about 50% with the water in the MPC. The dose at the center of the HI-TRAC is reduced by approximately 50% when there is also water in the water jacket and is essentially unchanged when there is no water in the water jacket as compared to the normal condition results shown in Table 5.1.7.

The burnup and cooling time combination of ~~42,500~~46,000 MWD/MTU and ~~5–3~~ years was selected for the 100-ton MPC-24 HI-TRAC analysis because this combination of burnup and cooling time results in the highest dose rates, and therefore, bounds all other requested combinations in the 100-ton HI-TRAC. For comparison, dose rates corresponding to a burnup of ~~52,500~~75,000 MWD/MTU and ~~4–5~~ year cooling time for the MPC-24 are provided in Table 5.4.4. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results clearly indicate that as the burnup and cooling time increase, the reduction in the gamma dose rate due to the increased cooling time results in a net decrease in the total dose rate. This result is due to the fact that the dose rates surrounding the 100-ton HI-TRAC transfer cask are gamma dominated.

In contrast, the dose rates surrounding the HI-TRAC 125 and 125D transfer casks have significantly higher neutron component. Therefore, the dose rates at ~~57,500~~75,000 MWD/MTU burnup and ~~12~~5 year cooling are slightly higher than the dose rates at ~~42,500~~46,000 MWD/MTU burnup and ~~5~~3 year cooling. The dose rates for the 125-ton HI-TRACs with the MPC-24 at ~~57,500~~75,000 MWD/MTU and ~~12~~5 year cooling are listed in Table 5.1.8 of Section 5.1. For comparison, dose rates corresponding to a burnup of ~~42,500~~46,000 MWD/MTU and ~~5~~3 year cooling time for the MPC-24 are provided in Table 5.4.5.

Tables 5.4.9 and 5.4.10 provide dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-68 at burnup and cooling time combinations of ~~40~~39,000 MWD/MTU and ~~5~~3 years and ~~50~~70,000 MWD/MTU and ~~10~~6 years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-68 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-68 at the center of the HI-TRAC, on-contact, and at locations ~~1 to 2 feet~~1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Tables 5.4.11 and 5.4.12 provide dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-32 at burnup and cooling time combinations of ~~32,500~~35,000 MWD/MTU and ~~5~~3 years and ~~45~~75,000 MWD/MTU and ~~10~~8 years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-32 case than in the MPC-24 case. However, the MPC-24 produces comparable or higher dose rates than the MPC-32 at the center of the HI-TRAC, on-contact, and at locations ~~1 to 2 feet~~1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

As mentioned in Section 5.0, all MPCs offer a regionalized loading pattern as described in ~~Appendix B to the CoC~~Section 2.1.9. This loading pattern authorizes fuel of higher decay heat than uniform loading (i.e. higher burnups and shorter cooling times) to be stored in the center region, region 1, of the MPC. The outer region, region 2, of the MPC in regionalized loading is authorized to store fuel of lower decay heat than uniform loading (i.e. lower burnups and longer cooling times). From a shielding perspective, the older fuel on the outside provides shielding for the inner fuel in the radial direction. Regionalized patterns were specifically analyzed in each MPC in the 100-ton HI-TRAC. Based on analysis using the same burnup and cooling times in region 1 and 2 the following percentages were calculated for dose location 2 on the 100-ton HI-TRAC.

- Approximately 21%, 27%, and 8% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24



- respectively. Region 1 contains 12 (38% of total), 32 (47% of total), and 4 (17% of total) assemblies in the MPC-32, MPC-68, and MPC-24 respectively.
- Approximately 1%, 2%, and 0.2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively.

These results clearly indicate that the outer fuel assemblies shield almost all of the gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. In the axial direction, regionalized loading results in higher dose rates in the center portion of the cask since the region 2 assemblies are not shielding the region 1 assemblies for axial dose locations.

~~All~~ *Bounding* burnup and cooling time combinations for regionalized loading were analyzed and compared to the dose rates from uniform loading patterns. It was concluded that, in general, the radial dose rates from regionalized loading are bounded by the radial dose rates from uniform loading patterns. Therefore, dose rates for specific regionalized loading patterns are not presented in this chapter. In the axial direction, the reverse may be true since the inner fuel assemblies in a regionalized loading pattern have a higher burnup than the assemblies in the uniform loading patterns. However, as depicted in the graphical data in Section 5.1.1, the dose rate along the pool or transfer lids decrease substantially moving radially outward from the center of the lid. Therefore, this increase in the dose rate in the center of the lids due to regionalized loading does not significantly impact the occupational exposure. Section 5.4.9 provides additional discussion on regionalized loading dose rates compared to uniform loading dose rates.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

#### 5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations

The HI-STORM 100 overpack and the HI-TRAC utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would

result in a localized reduction in the photon dose. In addition to the fins, the pocket trunnions in the HI-TRAC 100 and 125 are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Analysis of the pocket trunnions in the HI-TRAC 100 and 125 and the steel fins in the HI-TRAC 100, 125, and 125D indicate that neutron streaming is noticeable at the surface of the transfer cask. The neutron dose rate on the surface of the pocket trunnion is approximately 5 times higher than the circumferential average dose rate at that location. The gamma dose rate is approximately 10 times lower than the circumferential average dose rate at that location. The streaming at the rib location is the largest in the HI-TRAC 125D because the ribs are thicker than in the HI-TRAC 100 or 125. The neutron dose rate on the surface of the rib in the 125D is approximately 3 times higher than the circumferential average dose rate at that location. The gamma dose rate on the surface of the rib in the 125D is approximately 3 times lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins and the pocket trunnions compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask only surface average dose rates are reported in this chapter.

Below each lifting trunnion, there is a localized area where the water jacket has been reduced in height by 4.125 inches to accommodate the lift yoke (see Figures 5.3.12 and 5.3.13). This area experiences a significantly higher than average dose rate on contact of the HI-TRAC. The peak dose in this location is ~~1.52.6~~ 52.6 Rem/hr for the MPC-32, ~~1.41.9~~ 41.9 Rem/hr for the MPC-68 and ~~1.32.4~~ 32.4 Rem/hr for the MPC-24 in the 100-ton HI-TRAC and ~~649-1.7~~ 1.7 mrem/Rem/hr for the MPC-24 in the HI-TRAC 125D. At a distance of 1 to 2 feet from the edge of the HI-TRAC the localized effect is greatly reduced. This dose rate is acceptable because during lifting operations the lift yoke will be in place, which, due to the additional lift yoke steel (~3 inches), will greatly reduce the dose rate. However, more importantly, people will be prohibited from being in the vicinity of the lifting trunnions during lifting operations as a standard rigging practice. In addition the lift yoke is remote in its attachment and detachment, further minimizing personnel exposure. Immediately following the detachment of the lift yoke, in preparation for closure operations, temporary shielding may be placed in this area. Any temporary shielding (e.g., lead bricks, water tanks, lead blankets, steel plates, etc.) is sufficient to attenuate the localized hot spot. The operating procedure in Chapter 8 discusses the placement of temporary shielding in this area. For the 100-ton HI-TRAC, the optional temporary shield ring will replace the water that was lost from the axial reduction in the water jacket thereby eliminating the localized hot spot. When the HI-TRAC is in the horizontal position, during transport operations, it will (at a minimum) be positioned a few feet off the ground by the transport vehicle and therefore this location below the lifting trunnions will be positioned above people which will minimize the effect on personnel

exposure. In addition, good operating practice will dictate that personnel remain at least a few feet away from the transport vehicle. During vertical transport of a loaded HI-TRAC, the localized hot spot will be even further from the operating personnel. Based on these considerations, the conclusion is that this localized hot spot does not significantly impact the personnel exposure.

#### 5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

##### 5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed that the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Chapter 1, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Dividing the total fuel gamma source for a 6x6 fuel assembly in Table 5.2.7 by the 80 inch rubble height provides a gamma source per inch of  $3.41\text{E}+12$  photon/s. Dividing the total neutron source for a 6x6 fuel assembly in Table 5.2.18 by 80 inches provides a neutron source per inch of  $2.75\text{E}+05$  neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of  $1.08\text{E}+13$  photon/s and  $9.17\text{E}+05$  neutron/s, respectively, *for a burnup and cooling time of 40,000 MWD/MTU and 5 years*. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling in ~~Tables 5.2.6 and 5.2.17~~ by the active fuel length of 144 inches. Therefore, damaged Dresden 1 and Humboldt Bay fuel assemblies are bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates from Dresden 1 or Humboldt Bay fuel assemblies are provided as they are bounded by the intact fuel analysis.

##### 5.4.2.2 Generic PWR and BWR Damaged Fuel

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact.

As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was performed to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Only the 100-ton HI-TRAC was analyzed because it can be concluded that if the dose rate change is not significant for the 100-ton HI-TRAC then the change will not be significant for the 125-ton HI-TRACs or the HI-STORM overpacks.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density which is higher than the fuel density for an intact fuel assembly. If the damaged fuel assembly were to fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. This scenario was analyzed with MCNP-4A in a conservative bounding fashion to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 (4 peripheral locations in the MPC-24E or MPC-24EF) and the MPC-68 (16 peripheral locations) with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

Tables 5.4.13 and 5.4.14 provide the results for the MPC-24 and MPC-68, respectively. Only the radial dose rates are provided since the axial dose rates will not be significantly affected because the damaged fuel assemblies are located on the periphery of the baskets. A comparison of these results to the results in Tables 5.1.7 and 5.4.9 indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase by less than 20% while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the bottom and top is due to the assumed flat power distribution. The dose rates shown in Tables 5.4.13 and 5.4.14 were averaged over the circumference of the cask. Since almost all of the peripheral cells in the MPC-68 are filled with DFCs, an azimuthal variation would not be expected for the MPC-68. However, since there are only 4 DFCs in the MPC-24E, an azimuthal variation in dose due to the damaged fuel/fuel debris might be expected. Therefore, the dose rates were evaluated in four smaller regions, one outside each DFC, that encompass about 44% of the circumference. There was no significant change in the dose rate as a result of the localized dose calculation. These results indicate that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will

have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter.

*The MPC-32 was not explicitly analyzed for damaged fuel or fuel debris in this chapter. However, based on the analysis described above for the MPC-24 and the MPC-68, it can be concluded that the shielding performance of the MPC-32 will not be significantly affected by the storage of damaged fuel.*

#### 5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single HI-STORM overpack loaded with an MPC-24 and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters were evaluated with MCNP. In the model, the casks were placed on an infinite slab of dirt to account for earth-shine effects. The atmosphere was represented by dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 700 meters. This is more than sufficient to properly account for skyshine effects. The models included either 500 or 1050 meters of air around the cask. Based on the behavior of the dose rate as a function of distance, 50 meters of air, beyond the detector locations, is sufficient to account for back-scattering. Therefore, the HI-STORM MCNP off-site dose models account for back scattering by including more than 50 meters of air beyond the detector locations for all cited dose rates. Since gamma back-scattering has an effect on the off-site dose, it is recommended that the site-specific evaluation under 10CFR72.212 include at least 50 to 100 meters of air, beyond the detector locations, in the calculational models.

The MCNP calculations of the off-site dose used a two-stage process. In the first stage a binary surface source file (MCNP terminology) containing particle track information was written for particles crossing the outer radial and top surfaces of the HI-STORM overpack. In the second stage of the calculation, this surface source file was used with the particle tracks originating on the outer edge of the overpack and the dose rate was calculated at the desired location (hundreds of meters away from the overpack). The results from this two-stage process are statistically the same as the results from a single calculation. However, the advantage of the two-stage process is that each stage can be optimized independently.

The annual dose, assuming 100% occupancy (8760 hours), at ~~200-250~~ meters from one cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed. This table indicates that the dose due to neutrons is ~~7-2.5~~ % of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques

may not properly account for the neutron transmissions and could lead to low estimates of the site boundary dose.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = A.
2. The annual dose from the radiation leaving the top of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = B.
3. The annual dose from the radiation leaving the side of a HI-STORM 100 overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7 for the bounding burnup and cooling time of ~~5247,500~~ 5247.5 MWD/MTU and 53-year cooling. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM 100 overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

$$\text{Dose} = ZA + 2ZB + ZC$$

As an example, the dose from a 2x3 array at ~~300-400~~ 300-400 meters is presented.

1. The annual dose from the side of a single cask: Dose A = 4.38
2. The annual dose from the top of a single cask: Dose B = 1.65E-2
3. The annual dose from the side of a cask positioned behind another cask:  
Dose C = 0.88

Using the formula shown above (Z=3), the total dose at ~~300-400~~ 300-400 meters from a 2x3 array of HI-STORM overpacks is ~~19.11~~ 15.88 mrem/year, assuming a 8760 hour occupancy.

An important point to notice here is that the dose from the side of the back row of casks is *approximately* 16 % of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis. Therefore, a site-specific evaluation of dose at the controlled area boundary

must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

#### 5.4.4 Stainless Steel Clad Fuel Evaluation

Table 5.4.8 presents the dose rates at the center of the HI-STORM 100 overpack, adjacent and at one meter distance, from the stainless steel clad fuel. These dose rates, when compared to Tables 5.1.1 through 5.1.6, are similar to the dose rates from the design basis zircaloy clad fuel, indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STORM 100 overpack. Since the true active fuel length is shorter than 144 inches and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

#### 5.4.5 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the source terms for the BWR design basis zircaloy clad fuel assembly (GE 7x7) which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.22 by the 110 inch active fuel height provides a gamma source per inch of  $2.36\text{E}+12$  photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.23 by 110 inches provides a neutron source strength per inch of  $3.06\text{E}+5$  neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of  $1.08\text{E}+13$  photons/s and  $9.17\text{E}+5$  neutrons/s *for 40,000 MWD/MTU and 5 year cooling*. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling ~~in Tables 5.2.6 and 5.2.17~~ by the active fuel length of 144 inches. This comparison shows that the MOX fuel source

terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Section 5.4.2.1, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches. The resulting gamma and neutron source strengths are  $3.25\text{E}+12$  photons/s and  $4.21\text{E}+5$  neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

#### 5.4.6 Non-Fuel Hardware

As discussed in Section 5.2.4, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM 100 System. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. BPRAs and TPDs are authorized for unrestricted storage in an MPC while the CRAs and APSRs are restricted to the center four locations in the MPC-24, MPC-24E, MPC-24EF and MPC-32. The calculation of the source term and a description of the bounding fuel devices was provided in Section 5.2.4. The dose rate due to BPRAs and TPDs being stored in a fuel assembly was explicitly calculated. Table 5.4.15 provides the dose rates at various locations on the surface and one meter from the 100-ton HI-TRAC due to the BPRAs and TPDs for the MPC-24 and MPC-32. These results were added to the totals in the other table to provide the total dose rate with BPRAs. Table 5.4.15 indicates that the dose rates from BPRAs bound the dose rates from TPDs.

As discussed in Section 5.2.4, two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs. The dose rate due to CRAs and APSRs being stored in the inner four fuel locations was explicitly calculated for dose locations around the 100-ton HI-TRAC. Tables 5.4.16 and 5.4.17 provide the results for the different configurations of CRAs and APSRs, respectively, in the MPC-24 and MPC-32. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is minimal and the dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, as noted in Tables 5.4.16 and 5.4.17, the dose rate at the edge of the transfer lid is almost negligible due to APSRs and CRAs. Therefore, even though the dose rates calculated (using a very conservative source term evaluation) are daunting, they do not pose a risk from an operations perspective because they are localized in nature. Section 5.1.1 provides additional discussion on the acceptability of the relatively high localized doses on the bottom of the HI-TRACs.



#### 5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately  $1\text{E}+8$  neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as  $5.8\text{E}-6$  neutrons per gamma ( $1\text{E}+8/865/3.7\text{E}+10/0.54$ ) with energy greater than 1.666 MeV or  $1.16\text{E}+5$  neutrons/curie ( $1\text{E}+8/865$ ) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be  $1.04\text{E}+8$  gammas/sec which would produce a neutron source of 603.2 neutrons/sec ( $1.04\text{E}+8 * 5.8\text{E}-6$ ). 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  $4.63\text{E}+6$  neutrons/sec ( $39.9 * 1.16\text{E}+5$ ) or  $6.0\text{E}+4$  neutrons/sec/inch ( $4.63\text{E}+6/77.25$ ). 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 15% 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.

#### 5.4.9 Regionalized Loading Dose Rate Evaluation

Dose rates were calculated for regionalized loading patterns for the MPC-24, MPC-32, and MPC-68 using MCNP-4A. *Burnup and cooling time combinations bounding the 14x14A and 9x9G array classes were used in the analysis since for uniform loading these array classes have the highest permissible burnup for a given cooling time. Section 2.1.9 describes the calculation of the allowable burnup and cooling times for regionalized loading. Burnup and cooling time combinations which bound the regionalized values were analyzed in this section. All burnup and cooling time combinations in Appendix B to the CoC were analyzed for both uniform and regionalized loading.* The dose rates for all dose locations reported in this chapter were compared for the uniform loading patterns and the regionalized loading patterns.

It was determined that for the MPC-32, all radial ~~surface and~~ 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated ~~surface~~ dose rates in the axial locations for regionalized loading were less than ~~1510%~~ higher than the uniform dose rates reported in this chapter ~~for the surface at 1 meter from of the overpack. At one meter from the overpack, dose location 4 (in the center) was the only dose location which produced a slightly higher (5%) dose rate for regionalized loading compared to uniform loading.~~

For the MPC-24 and MPC-68 it was determined that *all 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. the maximum calculated dose rates in the axial direction for regionalized loading were less than 21% higher than the maximum calculated dose rates for uniform loading reported in this chapter. At one meter distance, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates. In the radial direction, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates for both surface and one meter locations.*

For the MPC-68 it was determined that all radial ~~surface and~~ 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated ~~surface~~ dose rates in the axial locations for regionalized loading were less than 21% higher than the uniform dose rates reported in this chapter for the surface of the overpack. At one meter from the overpack, dose locations 4 (in the center) and 5 (transfer lid center) were the only dose locations which produced a slightly higher (5% and 1.5% respectively) dose rate for regionalized loading compared to uniform loading.

Based on these results it can be stated that regionalized loading patterns will reduce the dose rate in the radial direction by shielding the hotter fuel on the inside of the cask with colder fuel on the outside of the cask. However, in the axial direction the localized dose rates in the center of the cask may increase as a result of the regionalized loading pattern. This is a localized effect, which has dissipated at the edge of the cask, and therefore will not result in a significant increase to the occupational exposure rates. In addition, it should be mentioned that the localized increase on

the bottom center of the overpack is an area where workers will normally not be present and the increase in the top center of the overpack is an area where workers minimize their stay.

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])

<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])

<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS  
(FROM [5.4.1])

Neutron Energy (MeV)	Quality Factor	(rem/hr) <sup>†</sup> /(n/cm <sup>2</sup> -s)
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

<sup>†</sup> Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC  
 CONDITION WITH AN EMPTY NEUTRON SHIELD  
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
~~42,500~~ 46,000 MWD/MTU AND 53-YEAR COOLING

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	34.70	282.04	24.34	341.07	343.69
2	2212.18	0.66	423.66	2636.50	2839.98
3	8.60	429.18	5.92	443.70	575.29
4	31.22	326.11	0.98	358.31	460.57
5 (pool lid)	111.45	1835.89	3.33	1950.68 <sup>†††</sup>	1960.87
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	294.71	63.68	60.22	418.60	445.03
2	979.53	6.23	139.30	1125.06	1215.29
3	117.27	104.40	24.91	246.57	290.89

Note: MPC internal water level is 10 inches below the MPC lid.

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.



Table 5.4.3

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC  
 CONDITION WITH A FULL NEUTRON SHIELD  
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
~~42,500~~46,000 MWD/MTU AND 53-YEAR COOLING

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	29.73	282.19	3.17	315.10	317.20
2	1313.86	0.44	27.67	1341.97	1457.53
3	5.61	428.14	0.56	434.31	565.24
4	31.19	326.10	1.00	358.30	460.55
5 (pool lid)	111.11	1836.07	2.82	1950.01 <sup>†††</sup>	1960.18
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	170.07	43.84	3.70	217.61	232.40
2	573.05	3.49	10.41	586.95	637.50
3	67.61	72.02	1.26	140.89	169.77

Note: MPC internal water level is 10 inches below the MPC lid.

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

<sup>†††</sup> Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.4

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
~~52,500~~75,000 MWD/MTU AND 105-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	61.51	59.98	841.88	848.87	1812.25	1820.79
2	1720.78	244.11	0.84	450.27	2416.00	2663.24
3	16.84	11.76	464.19	710.32	1203.11	1351.64
3 (temp)	7.62	20.93	215.15	11.42	255.11	323.26
4	41.62	4.64	373.59	874.50	1294.35	1418.85
4 (outer)	11.60	2.95	93.02	590.24	697.81	729.14
5 (pool lid)	298.84	85.64	4241.72	5701.99	10328.19	10392.96
5 (transfer)	732.62	4.69	6320.81	3264.99	10323.11	10419.95
5(t-outer)	178.17	1.60	611.80	1290.11	2081.69	2103.16
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	226.79	32.24	125.15	137.98	522.16	554.64
2	754.47	74.62	9.90	168.82	1007.81	1117.29
3	94.60	17.96	103.96	66.26	282.78	332.22
3 (temp)	94.09	19.29	88.55	25.04	226.97	271.54
4	14.19	0.81	115.34	217.83	348.17	386.74
5 (transfer)	315.47	0.86	2582.07	911.26	3809.66	3848.79
5(t-outer)	42.95	2.78	232.74	261.61	540.08	543.98

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.5

DOSE RATES FROM THE 125-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
42,500 46,000 MWD/MTU AND 53-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 125-TON HI-TRAC</b>						
1	12.43	17.84	101.50	119.96	251.73	252.44
2	211.88	52.83	0.01	83.09	347.81	363.68
3	2.66	1.89	62.80	191.39	258.73	278.44
4	71.16	2.42	343.61	221.50	638.70	754.12
4 (outer)	9.28	1.73	42.67	4.65	58.33	72.52
5 (pool)	108.22	0.90	529.32	766.89	1405.34	1413.04
5 (transfer)	112.43	1.38	606.59	127.01	847.40	852.89
<b>ONE METER FROM THE 125-TON HI-TRAC</b>						
1	28.53	7.12	13.01	19.74	68.40	70.43
2	95.52	17.13	0.53	28.34	141.51	148.58
3	10.94	4.02	12.69	17.61	45.26	50.18
4	20.01	0.58	82.73	22.81	126.13	153.78
5 (transfer)	41.40	0.27	293.26	22.00	356.92	359.85

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.6

ANNUAL DOSE AT ~~200-250~~ METERS FROM A SINGLE  
 HI-STORM OVERPACK WITH AN MPC-24 WITH DESIGN BASIS  
 ZIRCALOY CLAD FUEL<sup>†</sup>

<b>Dose Component</b>	<b>5247,500 MWD/MTU 53-Year Cooling (mrem/yr)</b>
Fuel gammas <sup>††</sup>	16.5221.02
<sup>60</sup> Co Gammas	2.171.24
Neutrons	1.500.57
Total	20.1922.83

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<sup>†</sup> 8760 hour annual occupancy is assumed.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM  
 VARIOUS ISFSI CONFIGURATIONS  
 5247,500 MWD/MTU AND 53-YEAR COOLING ZIRCALOY CLAD FUEL<sup>†</sup>

<b>Distance</b>	<b>A Side of Overpack (mrem/yr)</b>	<b>B Top of Overpack (mrem/yr)</b>	<b>C Side of Shielded Overpack (mrem/yr)</b>
100 meters	306.6	1.44	61.32
150 meters	109.2	0.56	21.84
200 meters	48.4	0.25	9.68
250 meters	24.0	0.11	4.80
300 meters	12.9	6.09E-2	2.58
350 meters	7.48	3.12E-2	1.50
400 meters	4.38	1.65E-2	0.88

<sup>†</sup> 8760 hour annual occupancy is assumed.

Table 5.4.8

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR  
DESIGN BASIS STAINLESS STEEL CLAD FUEL  
WITHOUT BPRAs

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>MPC-24 (40,000 MWD/MTU AND 8-YEAR COOLING)</b>				
2 (Adjacent)	36.97	0.02	1.11	38.10
2 (One Meter)	18.76	0.17	0.50	19.43
<b>MPC-32 (40,000 MWD/MTU AND 9-YEAR COOLING)</b>				
2 (Adjacent)	37.58	0.00	1.49	39.08
2 (One Meter)	18.74	0.25	0.58	19.57
<b>MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)</b>				
2 (Adjacent)	17.79	0.01	0.10	17.90
2 (One Meter)	8.98	0.13	0.04	9.15

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† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.9

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
4039,000 MWD/MTU AND 53-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	101.95	13.60	1148.49	181.72	1445.78
2	2235.38	67.10	0.73	121.56	2424.77
3	6.47	1.42	695.08	77.00	779.97
3 (temp)	3.74	2.27	330.06	1.42	337.49
4	14.60	0.60	273.56	100.40	389.15
4 (outer)	4.04	0.40	72.46	60.14	137.03
5 (pool lid)	302.21	16.66	5142.93	1089.34	6551.14
5 (transfer lid)	424.40	0.78	7748.87	686.26	8860.30
5 (t-outer)	157.68	0.33	683.21	256.32	1097.53
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	304.87	8.25	107.26	32.11	452.50
2	962.27	18.99	7.91	41.90	1031.07
3	75.11	3.26	157.28	8.89	244.54
3 (temp)	75.04	3.41	127.40	4.27	210.11
4	5.39	0.11	91.29	20.93	117.72
5 (transfer lid)	218.21	0.33	3437.93	183.88	3840.35
5 (t-outer)	27.48	0.59	290.36	51.99	370.41

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.10

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
5070,000 MWD/MTU AND 106-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	46.14	51.49	957.08	687.54	1742.24
2	1122.53	253.96	0.61	459.77	1836.86
3	2.37	5.36	579.24	291.33	878.29
3 (temp)	1.52	8.60	275.05	5.39	290.56
4	5.50	2.26	227.96	379.71	615.44
4 (outer)	1.59	1.52	60.38	227.53	291.02
5 (pool lid)	138.45	63.04	4285.78	4121.87	8609.13
5 (transfer lid)	239.62	2.94	6457.39	2597.01	9296.96
5 (t-outer)	81.20	1.24	569.34	969.87	1621.64
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	151.92	31.24	89.38	121.49	394.03
2	480.13	71.90	6.59	158.45	717.07
3	37.29	12.34	131.07	33.63	214.32
3 (temp)	37.24	12.90	106.17	16.14	172.44
4	2.35	0.43	76.08	79.17	158.02
5 (transfer lid)	116.13	1.25	2864.94	695.86	3678.18
5 (t-outer)	14.24	2.22	241.96	196.72	455.15

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.



Table 5.4.11

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL  
32,50035,000 MWD/MTU AND 53-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	101.70	7.68	943.95	111.78	1165.11	1175.39
2	2499.72	34.07	1.35	63.94	2599.08	2890.62
3	35.51	1.50	595.93	88.06	721.00	946.40
4	84.64	0.99	446.65	110.31	642.58	821.55
4 (outer)	23.54	0.39	112.05	75.06	211.03	256.03
5 (pool)	604.32	10.75	5208.45	727.74	6551.26	6633.59
5 (transfer)	1056.27	0.42	7860.47	408.03	9325.19	9424.04
5(t-outer)	199.21	0.23	662.48	162.60	1024.51	1044.60
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	334.53	4.51	140.68	18.47	498.20	536.18
2	1107.24	10.70	10.72	23.33	1151.99	1281.43
3	144.93	2.54	122.47	8.79	278.73	345.55
4	26.50	0.14	133.19	27.24	187.07	240.23
5 (transfer)	447.93	0.14	3124.65	113.90	3686.60	3730.23
5(t-outer)	46.85	0.43	277.06	33.17	357.51	361.97

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.12

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL  
4575,000 MWD/MTU AND 408-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	35.36	58.40	756.90	849.12	1699.78	1710.06
2	1008.77	258.96	1.09	485.41	1754.23	2045.77
3	10.68	11.39	477.85	669.04	1168.96	1394.36
4	29.28	7.51	358.14	837.70	1232.64	1411.61
4 (outer)	7.45	2.97	89.84	570.22	670.48	715.48
5 (pool)	239.46	81.70	4176.38	5528.60	10026.14	10108.46
5 (transfer)	464.10	3.16	6302.90	3101.39	9871.56	9970.41
5(t-outer)	82.86	1.72	531.20	1235.27	1851.06	1871.15
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	130.79	34.31	112.81	140.25	418.16	456.14
2	441.14	81.34	8.60	177.10	708.18	837.61
3	56.21	19.30	98.20	66.75	240.47	307.29
4	8.34	1.07	106.80	206.96	323.17	376.33
5 (transfer)	186.92	1.03	2505.49	865.27	3558.70	3602.33
5(t-outer)	18.96	3.25	222.16	251.98	496.36	500.83

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.13

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS  
 WITH FOUR DAMAGED FUEL CONTAINERS  
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL  
 42,50046,000 MWD/MTU AND 53-YEAR COOLING  
 WITHOUT BPRAs

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,γ) Gammas (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	139.14	20.90	849.14	317.12	1326.29
2	2635.40	75.69	1.01	137.79	2849.89
3	40.59	4.69	468.20	304.41	817.88
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	376.67	10.74	126.22	48.19	561.82
2	1175.03	23.42	9.99	51.90	1260.33
3	169.41	6.13	104.86	28.32	308.72

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.14

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS  
 WITH SIXTEEN DAMAGED FUEL CONTAINERS  
 MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL  
 4039,000 MWD/MTU AND 53-YEAR COOLING

<b>Dose Point<sup>†</sup> Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,γ) Gammas (mrem/hr)</b>	<b><sup>60</sup>Co Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
<b>ADJACENT TO THE 100-TON HI-TRAC</b>					
1	237.63	19.51	1148.49	336.58	1742.21
2	2107.52	67.58	0.73	114.34	2290.16
3	8.12	2.86	695.08	170.31	876.38
<b>ONE METER FROM THE 100-TON HI-TRAC</b>					
1	353.82	10.21	107.26	48.47	519.76
2	923.33	20.45	9.30	46.37	999.45
3	103.30	4.54	157.28	17.67	282.78

<sup>†</sup> Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.15

DOSE RATES DUE TO BPRAs AND TPDs FROM THE 100-TON HI-TRAC  
FOR NORMAL CONDITIONS

Dose Point Location	MPC-24		MPC-32	
	BPRAs (mrem/hr)	TPDs (mrem/hr)	BPRAs (mrem/hr)	TPDs (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>				
1	8.54	0.00	10.28	0.01
2	247.24	0.03	291.53	0.04
3	148.53	125.75	225.40	188.04
3 (temp)	68.15	56.21	93.60	76.97
4	124.50	106.71	178.97	156.15
4 (outer)	31.33	27.12	45.00	39.32
5 (pool lid)	64.77	0.00	82.32	0.00
5 (transfer lid)	96.84	0.00	98.85	0.00
5 (t-outer)	21.47	0.00	20.09	0.00
<b>ONE METER FROM THE 100-TON HI-TRAC</b>				
1	32.48	0.18	37.98	0.23
2	109.47	1.20	129.44	1.62
3	49.43	38.93	66.82	54.93
3 (temp)	44.57	35.01	59.10	48.77
4	38.57	33.37	53.16	47.19
5 (transfer lid)	39.13	0.00	43.62	0.00
5 (t-outer)	3.90	0.00	4.47	0.00

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.16

DOSE RATES DUE TO CRAs FROM THE 100-TON HI-TRAC  
FOR NORMAL CONDITIONS

Dose Point Location	MPC-24		MPC-32	
	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>				
1	5.39	1.02	3.28	0.68
2	0.09	0.00	0.01	0.00
3	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5 (pool lid)	919.59	170.85	1141.10	213.24
5(transfer lid)	1519.98	287.72	2012.93	380.57
5(t-outer)	1.54	0.25	1.01	0.19
<b>ONE METER FROM THE 100-TON HI-TRAC</b>				
1	1.20	0.20	0.69	0.14
2	0.26	0.03	0.05	0.01
3	0.01	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5(transfer lid)	223.62	41.60	257.95	49.19
5(t-outer)	8.26	1.54	8.87	1.70

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.17

DOSE RATES DUE TO APSRs FROM THE 100-TON HI-TRAC  
FOR NORMAL CONDITIONS

Dose Point Location	MPC-24			MPC-32		
	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)
<b>ADJACENT TO THE 100-TON HI-TRAC</b>						
1	12.42	2.35	12.25	7.57	1.56	7.51
2	0.21	0.01	9.12	0.03	0.00	0.19
3	0.00	0.00	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00	0.00	0.00
5 (pool lid)	1996.57	371.98	1941.51	2414.84	453.88	2687.17
5(transfer)	3021.08	572.85	2994.54	3980.02	750.17	3860.83
5(t-outer)	3.41	0.54	3.57	2.23	0.42	1.94
<b>ONE METER FROM THE 100-TON HI-TRAC</b>						
1	2.73	0.46	3.49	1.57	0.32	1.58
2	0.61	0.07	3.31	0.12	0.02	0.18
3	0.02	0.00	0.04	0.01	0.00	0.01
4	0.00	0.00	0.00	0.00	0.00	0.00
5(transfer)	458.06	84.81	444.44	521.02	99.10	510.78
5(t-outer)	17.11	3.19	17.36	18.34	3.48	18.20

## Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.18

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR  
DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

<b>Assembly</b>	<b>Active fuel length (inch)</b>	<b>Neutrons per sec per inch</b>	<b>Neutrons per sec per inch with Sb-Be source</b>	<b>Reference for neutrons per sec per inch</b>
7x7 design basis	144	9.17E+5	N/A	Table 5.2.17—40 GWD/MTU and 5 year cooling
6x6 design basis	110	2.0E+5	2.6E+5	Table 5.2.18
6x6 design basis MOX	110	3.06E+5	3.66E+5	Table 5.2.23



## 5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this FSAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

This chapter has evaluated these shielding SSCs important to safety and has assessed the impact on health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM 100 System.

It has been shown that the design of the shielding system of the HI-STORM 100 System is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel.

## 5.6 REFERENCES

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**APPENDIX 5.A**

**SAMPLE INPUT FILE FOR SAS2H**

**(Total number of pages in this appendix : 3)**

```

=SAS2H    PARM='halt05,skipshipdata'
bw 15x15 PWR assembly
' fuel temp 923
44groupndf5    LATTICECELL
UO2 1 0.95 923 92234 0.03204 92235 3.6 92236 0.01656
    92238 96.3514 END
,
' Zirc 4 composition
ARBM-ZIRC4 6.55 4 1 0 0 50000 1.7 26000 0.24 24000 0.13 40000 97.93
    2 1.0 595 END
,
' water with 652.5 ppm boron
H2O    3 DEN=0.7135 1 579 END
ARBM-BORMOD 0.7135 1 1 0 0 5000 100 3 652.5E-6 579 END
,
co-59  3 0 1-20 579 end
kr-83  1 0 1-20 923 end
kr-84  1 0 1-20 923 end
kr-85  1 0 1-20 923 end
kr-86  1 0 1-20 923 end
sr-90  1 0 1-20 923 end
y-89   1 0 1-20 923 end
zr-94  1 0 1-20 923 end
zr-95  1 0 1-20 923 end
mo-94  1 0 1-20 923 end
mo-95  1 0 1-20 923 end
nb-94  1 0 1-20 923 end
nb-95  1 0 1-20 923 end
tc-99  1 0 1-20 923 end
ru-106 1 0 1-20 923 end
rh-103 1 0 1-20 923 end
rh-105 1 0 1-20 923 end
sb-124 1 0 1-20 923 end
sn-126 1 0 1-20 923 end
xe-131 1 0 1-20 923 end
xe-132 1 0 1-20 923 end
xe-134 1 0 1-20 923 end
,
xe-135 1 0 1-09 923 end
,
xe-136 1 0 1-20 923 end
cs-133 1 0 1-20 923 end
cs-134 1 0 1-20 923 end
cs-135 1 0 1-20 923 end
cs-137 1 0 1-20 923 end
ba-136 1 0 1-20 923 end
la-139 1 0 1-20 923 end
ce-144 1 0 1-20 923 end
pr-143 1 0 1-20 923 end
nd-143 1 0 1-20 923 end
nd-144 1 0 1-20 923 end
nd-145 1 0 1-20 923 end
nd-146 1 0 1-20 923 end
nd-147 1 0 1-20 923 end
nd-148 1 0 1-20 923 end
nd-150 1 0 1-20 923 end
pm-147 1 0 1-20 923 end
pm-148 1 0 1-20 923 end
pm-149 1 0 1-20 923 end

```

```

sm-147  1 0 1-20 923 end
sm-148  1 0 1-20 923 end
sm-149  1 0 1-20 923 end
sm-150  1 0 1-20 923 end
sm-151  1 0 1-20 923 end
sm-152  1 0 1-20 923 end
eu-151  1 0 1-20 923 end
eu-153  1 0 1-20 923 end
eu-154  1 0 1-20 923 end
eu-155  1 0 1-20 923 end
gd-154  1 0 1-20 923 end
gd-155  1 0 1-20 923 end
gd-157  1 0 1-20 923 end
gd-158  1 0 1-20 923 end
gd-160  1 0 1-20 923 end
'
END COMP
'
' - - - - -
'
'   FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH  1.44272 0.950468 1 3 1.08712 2 0.97028 0 END
'
' - - - - -
'
'   MTU in this model is 0.495485 based on fuel dimensions provided
'
'   1 power cycle will be used and a library will be generated every
'   2500 MWD/MTU power level is 40 MW/MTU
'   therefore 62.5 days per 2500 MWD/MTU
'   Below
'   BURN=62.5*NLIB/CYC
'   POWER=MTU*40
'
'   Number of libraries is 20 which is 50,000 MWD/MTU burnup (20*2500)
'
'   ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=208 FUELNGTH=365.76 NCYCLES=1 NLIB/CYC=20
PRINTLEVEL=1
LIGHTTEL=5 INPLEVEL=1 NUMHOLES=17
NUMINStr= 0 ORTUBE= 0.6731 SRTUBE=0.63246 END
POWER=19.81938 BURN=1250.0 END
'
O 66.54421
FE 0.24240868
ZR 98.78151 CR 0.1311304 SN 1.714782
'
END
=SAS2H PARM='restarts, halt10, skipshipdata'
bw 15x15 PWR assembly
END
=SAS2H PARM='restarts, halt15, skipshipdata'
bw 15x15 PWR assembly
END
=SAS2H PARM='restarts, halt20, skipshipdata'
bw 15x15 PWR assembly
END

```

**APPENDIX 5.B**

**SAMPLE INPUT FILE FOR ORIGEN-S**

**(Total number of pages in this appendix : 7)**



```

#ORIGENS
0$$ A4 33 A8 26 A11 71 E
1$$ 1 T
bw 15x15 FUEL -- FT33F001 -
'
' SUBCASE 1 LIBRARY POSITION 1
'
' lib pos grms photon group
3$$ 33 A3 1 0 A16 2 E T
35$$ 0 T
56$$ 5 5 A6 3 A10 0 A13 9 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
FUEL 3.6
BW 15x15 0.495485 MTU
58** 19.81938 19.81938 19.81938 19.81938 19.81938
60** 1.0000 3.0000 15.0000 30.0000 62.5
66$$ A1 2 A5 2 A9 2 E
73$$ 922350 922340 922360 922380 80000 500000
260000 240000 400000
74** 17837.45 158.7533 82.05225 477406.4 66544.21 1714.782
242.0868 131.1304 98781.51
75$$ 2 2 2 2 4 4 4 4 4 T
'
' SUBCASE 2 LIBRARY POSITION 2
'
3$$ 33 A3 2 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 5 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 3 LIBRARY POSITION 3
'
3$$ 33 A3 3 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 4 LIBRARY POSITION 4
'
3$$ 33 A3 4 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
'

```

```

' SUBCASE 5 LIBRARY POSITION 5
'
3$$ 33 A3 5 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 6 LIBRARY POSITION 6
'
3$$ 33 A3 6 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 7 LIBRARY POSITION 7
'
3$$ 33 A3 7 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 8 LIBRARY POSITION 8
'
3$$ 33 A3 8 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 9 LIBRARY POSITION 9
'
3$$ 33 A3 9 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T

```

```

'
' SUBCASE 10 LIBRARY POSITION 10
'
3$$$ 33 A3 10 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 11 LIBRARY POSITION 11
'
3$$$ 33 A3 11 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 12 LIBRARY POSITION 12
'
3$$$ 33 A3 12 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 13 LIBRARY POSITION 13
'
3$$$ 33 A3 13 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$$ A1 2 A5 2 A9 2 E T
'
' SUBCASE 14 LIBRARY POSITION 14
'
3$$$ 33 A3 14 0 A16 2 A33 0 E T
35$$$ 0 T
56$$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5

```

```

66$$ A1 2 A5 2 A9 2 E T
|
| SUBCASE 15 LIBRARY POSITION 15
|
3$$ 33 A3 15 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
|
| SUBCASE 16 LIBRARY POSITION 16
|
3$$ 33 A3 16 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
|
| SUBCASE 17 LIBRARY POSITION 17
|
3$$ 33 A3 17 0 A16 2 A33 0 E T
35$$ 0 T
56$$ 3 3 A6 3 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
|
| SUBCASE 18 LIBRARY POSITION 18
|
3$$ 33 A3 18 A4 7 0 A16 2 A33 18 E T
35$$ 0 T
56$$ 3 3 A6 1 A10 3 A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.05556 E T
fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2 E T
|
| SUBCASE - decay
|
54$$ A8 1 E
56$$ 0 9 A6 1 A10 3 A14 3 A15 1 A19 1 E
57** 0.0 0 1.E-5 E T
fuel enrichment above
60** 0.5 0.75 1.0 4.0 8.0 12.0 24.0 48.0 96.0
61** F0.1
65$$

```

```

'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z T
,
' SUBCASE - decay
,
54$$$ A8 1 E
56$$$ 0 9 A6 1 A10 9 A14 4 A15 1 A19 1 E
57** 4.0 0 1.E-5 E T
  fuel enrichment above
60** 10.0 20.0 30.0 60.0 90.0 120.0 180.0 240.0 365.0
61** F0.1
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      0 0 0      1 0 0      3Z      6Z T
,
' SUBCASE - decay
,
54$$$ A8 0 E
56$$$ 0 9 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 1.0 0 1.E-5 E T
  fuel enrichment above
60** 1.5 3.0 4.0 5.0 6.0 7.0 8.0 9.0 10.0
61** F1.0e-5
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
81$$$ 2 0 26 1 E
82$$$ 0 2 2 2 2 2 2 2 2
83** 1.1E+7 8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6 7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4 2.0E+4 1.0E+4
84** 20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
,
,
,
,
,
,
,
,
,
' SUBCASE - decay
,
54$$$ A8 0 E
56$$$ 0 10 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 10.0 0 1.E-5 E T
  fuel enrichment above
60** 11.0 12.0 13.0 14.0 15.0 16.0 17.0 18.0 19.0 20.0
61** F1.0e-5
65$$$
'GRAM-ATOMS   GRAMS   CURIES   WATTS-ALL   WATTS-GAMMA
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z
      3Z      0 1 0      1 0 0      1 0 0      3Z      6Z

```



**APPENDIX 5.C**

**SAMPLE INPUT FILE FOR MCNP**

**(Total number of pages in this appendix : 61)**

message: outp=hs24c11o srctp=hs24c11s runtpe=hs24c11r  
mctal=hs24c11m wssa=hs24c11w rssa=pt001w

hs24c11

```
c
c   origin is 6 inches below mpc
c
c   only cells that contain material are split axially
c   importance splitting is not done in cells with 0 material
c
c   axial segmentation is at the following boundaries
c   615, 620, 420, 430, 445, 455, 675, 651, 652 ,653
c   654, 655, 656, 657, 680
c
c   universe 1
c
301  0          (-40:41:-42:43)  -400 u=1
302  0  37 -38      -12          400 -410 u=1
303  0  37 -38    15           400 -410 u=1
304  0  35 -36      -20        400 -410 u=1
305  0  35 -36        23        400 -410 u=1
306  0  37 -38      -12          435 -460 u=1
307  0  37 -38    15           435 -460 u=1
308  0  35 -36      -20        435 -460 u=1
309  0  35 -36        23        435 -460 u=1
310  0  37 -38      -10          410 -435 u=1
311  0  37 -38    17           410 -435 u=1
312  0  35 -36      -18        410 -435 u=1
313  0  35 -36        25        410 -435 u=1
314  5 -7.92     10 -11 26 -27 410 -420 u=-1 $ left
315  6 -2.644    11 -12 26 -27 410 -420 u=-1 $ left
316  6 -2.644    15 -16 26 -27 410 -420 u=-1 $ right
317  5 -7.92     16 -17 26 -27 410 -420 u=-1 $ right
318  5 -7.92     28 -29 18 -19 410 -420 u=-1 $ bot
319  6 -2.644    28 -29 19 -20 410 -420 u=-1 $ bot
320  6 -2.644    28 -29 23 -24 410 -420 u=-1 $ top
321  5 -7.92     28 -29 24 -25 410 -420 u=-1 $ top
322  5 -7.92     10 -11 26 -27 420 -430 u=-1 $ left
323  6 -2.644    11 -12 26 -27 420 -430 u=-1 $ left
324  6 -2.644    15 -16 26 -27 420 -430 u=-1 $ right
325  5 -7.92     16 -17 26 -27 420 -430 u=-1 $ right
326  5 -7.92     28 -29 18 -19 420 -430 u=-1 $ bot
327  6 -2.644    28 -29 19 -20 420 -430 u=-1 $ bot
328  6 -2.644    28 -29 23 -24 420 -430 u=-1 $ top
329  5 -7.92     28 -29 24 -25 420 -430 u=-1 $ top
330  5 -7.92     10 -11 26 -27 430 -435 u=-1 $ left
331  6 -2.644    11 -12 26 -27 430 -435 u=-1 $ left
332  6 -2.644    15 -16 26 -27 430 -435 u=-1 $ right
333  5 -7.92     16 -17 26 -27 430 -435 u=-1 $ right
334  5 -7.92     28 -29 18 -19 430 -435 u=-1 $ bot
335  6 -2.644    28 -29 19 -20 430 -435 u=-1 $ bot
336  6 -2.644    28 -29 23 -24 430 -435 u=-1 $ top
337  5 -7.92     28 -29 24 -25 430 -435 u=-1 $ top
338  0           10 -12 27 -38 410 -435 u=-1 $ left
339  0           10 -12 37 -26 410 -435 u=-1 $ left
340  0           15 -17 27 -38 410 -435 u=-1 $ right
341  0           15 -17 37 -26 410 -435 u=-1 $ right
342  0           35 -28 18 -20 410 -435 u=-1 $ bot
343  0           29 -36 18 -20 410 -435 u=-1 $ bot
344  0           35 -28 23 -25 410 -435 u=-1 $ top
345  0           29 -36 23 -25 410 -435 u=-1 $ top
346  5 -7.92     12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-1
347  5 -7.92     12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-1
348  5 -7.92     12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-1
349  5 -7.92     12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-1
```



```

350 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-1
351 9 -1.17e-3 460 u=1
c fuel element
352 0 40 -41 42 -43 -415 u=1
353 5 -1.0783 40 -41 42 -43 415 -420 u=-1 $ lower nozzle
354 0 40 -41 42 -43 420 -425 u=-1 $ space
355 2 -3.8699 40 -41 42 -43 425 -430 u=-1 $ active fuel
356 5 -0.1591 40 -41 42 -43 430 -440 u=-1 $ space
357 5 -0.1591 40 -41 42 -43 440 -445 u=-1 $ plenum spacer
358 5 -1.5410 40 -41 42 -43 445 -455 u=-1 $ top nozzle
359 0 13 -14 21 -22 455 -460 u=-1
c
360 5 -7.92 38 -23 -12 400 -420 u=1
361 5 -7.92 20 -37 -12 400 -420 u=1
362 5 -7.92 12 -35 23 400 -420 u=1
363 5 -7.92 12 -35 -20 400 -420 u=1
364 5 -7.92 36 -15 23 400 -420 u=1
365 5 -7.92 36 -15 -20 400 -420 u=1
366 5 -7.92 38 -23 15 400 -420 u=1
367 5 -7.92 20 -37 15 400 -420 u=1
368 5 -7.92 38 -23 -12 420 -430 u=1
369 5 -7.92 20 -37 -12 420 -430 u=1
370 5 -7.92 12 -35 23 420 -430 u=1
371 5 -7.92 12 -35 -20 420 -430 u=1
372 5 -7.92 36 -15 23 420 -430 u=1
373 5 -7.92 36 -15 -20 420 -430 u=1
374 5 -7.92 38 -23 15 420 -430 u=1
375 5 -7.92 20 -37 15 420 -430 u=1
376 5 -7.92 38 -23 -12 430 -445 u=1
377 5 -7.92 20 -37 -12 430 -445 u=1
378 5 -7.92 12 -35 23 430 -445 u=1
379 5 -7.92 12 -35 -20 430 -445 u=1
380 5 -7.92 36 -15 23 430 -445 u=1
381 5 -7.92 36 -15 -20 430 -445 u=1
382 5 -7.92 38 -23 15 430 -445 u=1
383 5 -7.92 20 -37 15 430 -445 u=1
384 5 -7.92 38 -23 -12 445 -460 u=1
385 5 -7.92 20 -37 -12 445 -460 u=1
386 5 -7.92 12 -35 23 445 -460 u=1
387 5 -7.92 12 -35 -20 445 -460 u=1
388 5 -7.92 36 -15 23 445 -460 u=1
389 5 -7.92 36 -15 -20 445 -460 u=1
390 5 -7.92 38 -23 15 445 -460 u=1
391 5 -7.92 20 -37 15 445 -460 u=1
392 0 23 -12 400 -460 u=1
393 0 23 15 400 -460 u=1
394 0 15 -20 400 -460 u=1
395 0 -12 -20 400 -460 u=1
c
c universe 2
c
401 0 (-40:41:-42:43) -400 u=2
402 0 37 -38 -12 400 -410 u=2
403 0 37 -38 15 400 -410 u=2
404 0 35 -36 -20 400 -410 u=2
405 0 35 -36 23 400 -410 u=2
406 0 37 -38 -12 435 -460 u=2
407 0 37 -38 15 435 -460 u=2
408 0 35 -36 -20 435 -460 u=2
409 0 35 -36 23 435 -460 u=2
410 0 37 -38 -10 410 -435 u=2
411 0 37 -38 17 410 -435 u=2
412 0 35 -36 -18 410 -435 u=2
413 0 35 -36 25 410 -435 u=2
414 5 -7.92 10 -11 26 -27 410 -420 u=-2 $ left

```

```

415 6 -2.644 11 -12 26 -27 410 -420 u=-2 $ left
416 6 -2.644 15 -16 30 -31 410 -420 u=-2 $ right
417 5 -7.92 16 -17 30 -31 410 -420 u=-2 $ right
418 5 -7.92 28 -29 18 -19 410 -420 u=-2 $ bot
419 6 -2.644 28 -29 19 -20 410 -420 u=-2 $ bot
420 6 -2.644 32 -33 23 -24 410 -420 u=-2 $ top
421 5 -7.92 32 -33 24 -25 410 -420 u=-2 $ top
422 5 -7.92 10 -11 26 -27 420 -430 u=-2 $ left
423 6 -2.644 11 -12 26 -27 420 -430 u=-2 $ left
424 6 -2.644 15 -16 30 -31 420 -430 u=-2 $ right
425 5 -7.92 16 -17 30 -31 420 -430 u=-2 $ right
426 5 -7.92 28 -29 18 -19 420 -430 u=-2 $ bot
427 6 -2.644 28 -29 19 -20 420 -430 u=-2 $ bot
428 6 -2.644 32 -33 23 -24 420 -430 u=-2 $ top
429 5 -7.92 32 -33 24 -25 420 -430 u=-2 $ top
430 5 -7.92 10 -11 26 -27 430 -435 u=-2 $ left
431 6 -2.644 11 -12 26 -27 430 -435 u=-2 $ left
432 6 -2.644 15 -16 30 -31 430 -435 u=-2 $ right
433 5 -7.92 16 -17 30 -31 430 -435 u=-2 $ right
434 5 -7.92 28 -29 18 -19 430 -435 u=-2 $ bot
435 6 -2.644 28 -29 19 -20 430 -435 u=-2 $ bot
436 6 -2.644 32 -33 23 -24 430 -435 u=-2 $ top
437 5 -7.92 32 -33 24 -25 430 -435 u=-2 $ top
438 0 10 -12 27 -38 410 -435 u=-2 $ left
439 0 10 -12 37 -26 410 -435 u=-2 $ left
440 0 15 -17 31 -38 410 -435 u=-2 $ right
441 0 15 -17 37 -30 410 -435 u=-2 $ right
442 0 35 -28 18 -20 410 -435 u=-2 $ bot
443 0 29 -36 18 -20 410 -435 u=-2 $ bot
444 0 35 -32 23 -25 410 -435 u=-2 $ top
445 0 33 -36 23 -25 410 -435 u=-2 $ top
446 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-2
447 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-2
448 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-2
449 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-2
450 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-2
451 9 -1.17e-3 460 u=2
c
fuel element
452 0 40 -41 42 -43 -415 u=2
453 5 -1.0783 40 -41 42 -43 415 -420 u=-2 $ lower nozzle
454 0 40 -41 42 -43 420 -425 u=-2 $ space
455 2 -3.8699 40 -41 42 -43 425 -430 u=-2 $ active fuel
456 5 -0.1591 40 -41 42 -43 430 -440 u=-2 $ space
457 5 -0.1591 40 -41 42 -43 440 -445 u=-2 $ plenum spacer
458 5 -1.5410 40 -41 42 -43 445 -455 u=-2 $ top nozzle
459 0 13 -14 21 -22 455 -460 u=-2
c
460 5 -7.92 38 -23 -12 400 -420 u=2
461 5 -7.92 20 -37 -12 400 -420 u=2
462 0 12 -35 23 400 -420 u=2
463 5 -7.92 12 -35 -20 400 -420 u=2
464 0 36 -15 23 400 -420 u=2
465 5 -7.92 36 -15 -20 400 -420 u=2
466 0 38 -23 15 400 -420 u=2
467 0 20 -37 15 400 -420 u=2
468 5 -7.92 38 -23 -12 420 -430 u=2
469 5 -7.92 20 -37 -12 420 -430 u=2
470 0 12 -35 23 420 -430 u=2
471 5 -7.92 12 -35 -20 420 -430 u=2
472 0 36 -15 23 420 -430 u=2
473 5 -7.92 36 -15 -20 420 -430 u=2
474 0 38 -23 15 420 -430 u=2
475 0 20 -37 15 420 -430 u=2
476 5 -7.92 38 -23 -12 430 -445 u=2
477 5 -7.92 20 -37 -12 430 -445 u=2

```

478	0		12	-35	23	430	-445	u=2	
479	5	-7.92	12	-35	-20	430	-445	u=2	
480	0		36	-15	23	430	-445	u=2	
481	5	-7.92	36	-15	-20	430	-445	u=2	
482	0		38	-23	15	430	-445	u=2	
483	0		20	-37	15	430	-445	u=2	
484	5	-7.92	38	-23	-12	445	-460	u=2	
485	5	-7.92	20	-37	-12	445	-460	u=2	
486	0		12	-35	23	445	-460	u=2	
487	5	-7.92	12	-35	-20	445	-460	u=2	
488	0		36	-15	23	445	-460	u=2	
489	5	-7.92	36	-15	-20	445	-460	u=2	
490	0		38	-23	15	445	-460	u=2	
491	0		20	-37	15	445	-460	u=2	
492	0		23	-12		400	-460	u=2	
493	0		23	15		400	-460	u=2	
494	0		15	-20		400	-460	u=2	
495	0		-12	-20		400	-460	u=2	
c									
c	universe 3								
c									
501	0					(-40:41:-42:43)	-400	u=3	
502	0	37	-38			-12	400	-410	u=3
503	0	37	-38	15			400	-410	u=3
504	0	35	-36			-20	400	-410	u=3
505	0	35	-36		23		400	-410	u=3
506	0	37	-38		-12		435	-460	u=3
507	0	37	-38	15			435	-460	u=3
508	0	35	-36			-20	435	-460	u=3
509	0	35	-36		23		435	-460	u=3
510	0	37	-38		-10		410	-435	u=3
511	0	37	-38	17			410	-435	u=3
512	0	35	-36			-18	410	-435	u=3
513	0	35	-36		25		410	-435	u=3
514	5	-7.92		10	-11	30	-31	410	-420 u=-3 \$ left
515	6	-2.644		11	-12	30	-31	410	-420 u=-3 \$ left
516	6	-2.644		15	-16	26	-27	410	-420 u=-3 \$ right
517	5	-7.92		16	-17	26	-27	410	-420 u=-3 \$ right
518	5	-7.92		28	-29	18	-19	410	-420 u=-3 \$ bot
519	6	-2.644		28	-29	19	-20	410	-420 u=-3 \$ bot
520	6	-2.644		32	-33	23	-24	410	-420 u=-3 \$ top
521	5	-7.92		32	-33	24	-25	410	-420 u=-3 \$ top
522	5	-7.92		10	-11	30	-31	420	-430 u=-3 \$ left
523	6	-2.644		11	-12	30	-31	420	-430 u=-3 \$ left
524	6	-2.644		15	-16	26	-27	420	-430 u=-3 \$ right
525	5	-7.92		16	-17	26	-27	420	-430 u=-3 \$ right
526	5	-7.92		28	-29	18	-19	420	-430 u=-3 \$ bot
527	6	-2.644		28	-29	19	-20	420	-430 u=-3 \$ bot
528	6	-2.644		32	-33	23	-24	420	-430 u=-3 \$ top
529	5	-7.92		32	-33	24	-25	420	-430 u=-3 \$ top
530	5	-7.92		10	-11	30	-31	430	-435 u=-3 \$ left
531	6	-2.644		11	-12	30	-31	430	-435 u=-3 \$ left
532	6	-2.644		15	-16	26	-27	430	-435 u=-3 \$ right
533	5	-7.92		16	-17	26	-27	430	-435 u=-3 \$ right
534	5	-7.92		28	-29	18	-19	430	-435 u=-3 \$ bot
535	6	-2.644		28	-29	19	-20	430	-435 u=-3 \$ bot
536	6	-2.644		32	-33	23	-24	430	-435 u=-3 \$ top
537	5	-7.92		32	-33	24	-25	430	-435 u=-3 \$ top
538	0			10	-12	31	-38	410	-435 u=-3 \$ left
539	0			10	-12	37	-30	410	-435 u=-3 \$ left
540	0			15	-17	27	-38	410	-435 u=-3 \$ right
541	0			15	-17	37	-26	410	-435 u=-3 \$ right
542	0			35	-28	18	-20	410	-435 u=-3 \$ bot
543	0			29	-36	18	-20	410	-435 u=-3 \$ bot
544	0			35	-32	23	-25	410	-435 u=-3 \$ top

```

545 0 33 -36 23 -25 410 -435 u=-3 $ top
546 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-3
547 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-3
548 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-3
549 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-3
550 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-3
551 9 -1.17e-3 460 u=3
c
fuel element
552 0 40 -41 42 -43 -415 u=3
553 5 -1.0783 40 -41 42 -43 415 -420 u=-3 $ lower nozzle
554 0 40 -41 42 -43 420 -425 u=-3 $ space
555 2 -3.8699 40 -41 42 -43 425 -430 u=-3 $ active fuel
556 5 -0.1591 40 -41 42 -43 430 -440 u=-3 $ space
557 5 -0.1591 40 -41 42 -43 440 -445 u=-3 $ plenum spacer
558 5 -1.5410 40 -41 42 -43 445 -455 u=-3 $ top nozzle
559 0 13 -14 21 -22 455 -460 u=-3
c
560 0 38 -23 -12 400 -420 u=3
561 0 20 -37 -12 400 -420 u=3
562 0 12 -35 23 400 -420 u=3
563 5 -7.92 12 -35 -20 400 -420 u=3
564 0 36 -15 23 400 -420 u=3
565 5 -7.92 36 -15 -20 400 -420 u=3
566 5 -7.92 38 -23 15 400 -420 u=3
567 5 -7.92 20 -37 15 400 -420 u=3
568 5 -7.92 38 -23 -12 420 -430 u=3
569 5 -7.92 20 -37 -12 420 -430 u=3
570 0 12 -35 23 420 -430 u=3
571 0 12 -35 -20 420 -430 u=3
572 5 -7.92 36 -15 23 420 -430 u=3
573 0 36 -15 -20 420 -430 u=3
574 5 -7.92 38 -23 15 420 -430 u=3
575 5 -7.92 20 -37 15 420 -430 u=3
576 5 -7.92 38 -23 -12 430 -445 u=3
577 5 -7.92 20 -37 -12 430 -445 u=3
578 0 12 -35 23 430 -445 u=3
579 0 12 -35 -20 430 -445 u=3
580 5 -7.92 36 -15 23 430 -445 u=3
581 0 36 -15 -20 430 -445 u=3
582 5 -7.92 38 -23 15 430 -445 u=3
583 5 -7.92 20 -37 15 430 -445 u=3
584 5 -7.92 38 -23 -12 445 -460 u=3
585 5 -7.92 20 -37 -12 445 -460 u=3
586 0 12 -35 23 445 -460 u=3
587 0 12 -35 -20 445 -460 u=3
588 5 -7.92 36 -15 23 445 -460 u=3
589 0 36 -15 -20 445 -460 u=3
590 5 -7.92 38 -23 15 445 -460 u=3
591 5 -7.92 20 -37 15 445 -460 u=3
592 0 23 -12 400 -460 u=3
593 0 23 15 400 -460 u=3
594 0 15 -20 400 -460 u=3
595 0 -12 -20 400 -460 u=3
c
c universe 4
c
601 0 (-40:41:-42:43) -400 u=4
602 0 37 -38 -12 400 -410 u=4
603 0 37 -38 15 400 -410 u=4
604 0 35 -36 -20 400 -410 u=4
605 0 35 -36 23 400 -410 u=4
606 0 37 -38 -12 435 -460 u=4
607 0 37 -38 15 435 -460 u=4
608 0 35 -36 -20 435 -460 u=4
609 0 35 -36 23 435 -460 u=4

```

```

610 0 37 -38 -10 410 -435 u=4
611 0 37 -38 17 410 -435 u=4
612 0 35 -36 -18 410 -435 u=4
613 0 35 -36 25 410 -435 u=4
614 5 -7.92 10 -11 30 -31 410 -420 u=-4 $ left
615 6 -2.644 11 -12 30 -31 410 -420 u=-4 $ left
616 6 -2.644 15 -16 26 -27 410 -420 u=-4 $ right
617 5 -7.92 16 -17 26 -27 410 -420 u=-4 $ right
618 5 -7.92 32 -33 18 -19 410 -420 u=-4 $ bot
619 6 -2.644 32 -33 19 -20 410 -420 u=-4 $ bot
620 6 -2.644 28 -29 23 -24 410 -420 u=-4 $ top
621 5 -7.92 28 -29 24 -25 410 -420 u=-4 $ top
622 5 -7.92 10 -11 30 -31 420 -430 u=-4 $ left
623 6 -2.644 11 -12 30 -31 420 -430 u=-4 $ left
624 6 -2.644 15 -16 26 -27 420 -430 u=-4 $ right
625 5 -7.92 16 -17 26 -27 420 -430 u=-4 $ right
626 5 -7.92 32 -33 18 -19 420 -430 u=-4 $ bot
627 6 -2.644 32 -33 19 -20 420 -430 u=-4 $ bot
628 6 -2.644 28 -29 23 -24 420 -430 u=-4 $ top
629 5 -7.92 28 -29 24 -25 420 -430 u=-4 $ top
630 5 -7.92 10 -11 30 -31 430 -435 u=-4 $ left
631 6 -2.644 11 -12 30 -31 430 -435 u=-4 $ left
632 6 -2.644 15 -16 26 -27 430 -435 u=-4 $ right
633 5 -7.92 16 -17 26 -27 430 -435 u=-4 $ right
634 5 -7.92 32 -33 18 -19 430 -435 u=-4 $ bot
635 6 -2.644 32 -33 19 -20 430 -435 u=-4 $ bot
636 6 -2.644 28 -29 23 -24 430 -435 u=-4 $ top
637 5 -7.92 28 -29 24 -25 430 -435 u=-4 $ top
638 0 10 -12 31 -38 410 -435 u=-4 $ left
639 0 10 -12 37 -30 410 -435 u=-4 $ left
640 0 15 -17 27 -38 410 -435 u=-4 $ right
641 0 15 -17 37 -26 410 -435 u=-4 $ right
642 0 35 -32 18 -20 410 -435 u=-4 $ bot
643 0 33 -36 18 -20 410 -435 u=-4 $ bot
644 0 35 -28 23 -25 410 -435 u=-4 $ top
645 0 29 -36 23 -25 410 -435 u=-4 $ top
646 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-4
647 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-4
648 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-4
649 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-4
650 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-4
651 9 -1.17e-3 460 u=4
c
fuel element
652 0 40 -41 42 -43 -415 u=4
653 5 -1.0783 40 -41 42 -43 415 -420 u=-4 $ lower nozzle
654 0 40 -41 42 -43 420 -425 u=-4 $ space
655 2 -3.8699 40 -41 42 -43 425 -430 u=-4 $ active fuel
656 5 -0.1591 40 -41 42 -43 430 -440 u=-4 $ space
657 5 -0.1591 40 -41 42 -43 440 -445 u=-4 $ plenum spacer
658 5 -1.5410 40 -41 42 -43 445 -455 u=-4 $ top nozzle
659 0 13 -14 21 -22 455 -460 u=-4
c
660 0 38 -23 -12 400 -420 u=4
661 0 20 -37 -12 400 -420 u=4
662 5 -7.92 12 -35 23 400 -420 u=4
663 0 12 -35 -20 400 -420 u=4
664 5 -7.92 36 -15 23 400 -420 u=4
665 0 36 -15 -20 400 -420 u=4
666 5 -7.92 38 -23 15 400 -420 u=4
667 5 -7.92 20 -37 15 400 -420 u=4
668 0 38 -23 -12 420 -430 u=4
669 0 20 -37 -12 420 -430 u=4
670 5 -7.92 12 -35 23 420 -430 u=4
671 0 12 -35 -20 420 -430 u=4
672 5 -7.92 36 -15 23 420 -430 u=4

```

673	0		36	-15	-20	420	-430	u=4	
674	5	-7.92	38	-23	15	420	-430	u=4	
675	5	-7.92	20	-37	15	420	-430	u=4	
676	0		38	-23	-12	430	-445	u=4	
677	0		20	-37	-12	430	-445	u=4	
678	5	-7.92	12	-35	23	430	-445	u=4	
679	0		12	-35	-20	430	-445	u=4	
680	5	-7.92	36	-15	23	430	-445	u=4	
681	0		36	-15	-20	430	-445	u=4	
682	5	-7.92	38	-23	15	430	-445	u=4	
683	5	-7.92	20	-37	15	430	-445	u=4	
684	0		38	-23	-12	445	-460	u=4	
685	0		20	-37	-12	445	-460	u=4	
686	5	-7.92	12	-35	23	445	-460	u=4	
687	0		12	-35	-20	445	-460	u=4	
688	5	-7.92	36	-15	23	445	-460	u=4	
689	0		36	-15	-20	445	-460	u=4	
690	5	-7.92	38	-23	15	445	-460	u=4	
691	5	-7.92	20	-37	15	445	-460	u=4	
692	0		23	-12		400	-460	u=4	
693	0		23	15		400	-460	u=4	
694	0		15	-20		400	-460	u=4	
695	0		-12	-20		400	-460	u=4	
c									
c	universe 5								
c									
701	0					(-40:41:-42:43)	-400	u=5	
702	0	37	-38			-12	400	-410	u=5
703	0	37	-38	15			400	-410	u=5
704	0	35	-36			-20	400	-410	u=5
705	0	35	-36			23	400	-410	u=5
706	0	37	-38			-12	435	-460	u=5
707	0	37	-38	15			435	-460	u=5
708	0	35	-36			-20	435	-460	u=5
709	0	35	-36			23	435	-460	u=5
710	0	37	-38			-10	410	-435	u=5
711	0	37	-38	17			410	-435	u=5
712	0	35	-36			-18	410	-435	u=5
713	0	35	-36			25	410	-435	u=5
714	5	-7.92		10	-11	26	-27	410	-420 u=-5 \$ left
715	6	-2.644		11	-12	26	-27	410	-420 u=-5 \$ left
716	6	-2.644		15	-16	30	-31	410	-420 u=-5 \$ right
717	5	-7.92		16	-17	30	-31	410	-420 u=-5 \$ right
718	5	-7.92		32	-33	18	-19	410	-420 u=-5 \$ bot
719	6	-2.644		32	-33	19	-20	410	-420 u=-5 \$ bot
720	6	-2.644		28	-29	23	-24	410	-420 u=-5 \$ top
721	5	-7.92		28	-29	24	-25	410	-420 u=-5 \$ top
722	5	-7.92		10	-11	26	-27	420	-430 u=-5 \$ left
723	6	-2.644		11	-12	26	-27	420	-430 u=-5 \$ left
724	6	-2.644		15	-16	30	-31	420	-430 u=-5 \$ right
725	5	-7.92		16	-17	30	-31	420	-430 u=-5 \$ right
726	5	-7.92		32	-33	18	-19	420	-430 u=-5 \$ bot
727	6	-2.644		32	-33	19	-20	420	-430 u=-5 \$ bot
728	6	-2.644		28	-29	23	-24	420	-430 u=-5 \$ top
729	5	-7.92		28	-29	24	-25	420	-430 u=-5 \$ top
730	5	-7.92		10	-11	26	-27	430	-435 u=-5 \$ left
731	6	-2.644		11	-12	26	-27	430	-435 u=-5 \$ left
732	6	-2.644		15	-16	30	-31	430	-435 u=-5 \$ right
733	5	-7.92		16	-17	30	-31	430	-435 u=-5 \$ right
734	5	-7.92		32	-33	18	-19	430	-435 u=-5 \$ bot
735	6	-2.644		32	-33	19	-20	430	-435 u=-5 \$ bot
736	6	-2.644		28	-29	23	-24	430	-435 u=-5 \$ top
737	5	-7.92		28	-29	24	-25	430	-435 u=-5 \$ top
738	0			10	-12	27	-38	410	-435 u=-5 \$ left
739	0			10	-12	37	-26	410	-435 u=-5 \$ left

```

740 0 15 -17 31 -38 410 -435 u=-5 $ right
741 0 15 -17 37 -30 410 -435 u=-5 $ right
742 0 35 -32 18 -20 410 -435 u=-5 $ bot
743 0 33 -36 18 -20 410 -435 u=-5 $ bot
744 0 35 -28 23 -25 410 -435 u=-5 $ top
745 0 29 -36 23 -25 410 -435 u=-5 $ top
746 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 400 -420 u=-5
747 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 420 -430 u=-5
748 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 430 -445 u=-5
749 5 -7.92 12 -15 20 -23 (-13:-21:22:14) 445 -460 u=-5
750 0 13 -14 21 -22 (-40:41:-42:43) 400 -455 u=-5
751 9 -1.17e-3 460 u=5
c
c fuel element
752 0 40 -41 42 -43 -415 u=5
753 5 -1.0783 40 -41 42 -43 415 -420 u=-5 $ lower nozzle
754 0 40 -41 42 -43 420 -425 u=-5 $ space
755 2 -3.8699 40 -41 42 -43 425 -430 u=-5 $ active fuel
756 5 -0.1591 40 -41 42 -43 430 -440 u=-5 $ space
757 5 -0.1591 40 -41 42 -43 440 -445 u=-5 $ plenum spacer
758 5 -1.5410 40 -41 42 -43 445 -455 u=-5 $ top nozzle
759 0 13 -14 21 -22 455 -460 u=-5
c
760 5 -7.92 38 -23 -12 400 -420 u=5
761 5 -7.92 20 -37 -12 400 -420 u=5
762 5 -7.92 12 -35 23 400 -420 u=5
763 0 12 -35 -20 400 -420 u=5
764 5 -7.92 36 -15 23 400 -420 u=5
765 0 36 -15 -20 400 -420 u=5
766 0 38 -23 15 400 -420 u=5
767 0 20 -37 15 400 -420 u=5
768 5 -7.92 38 -23 -12 420 -430 u=5
769 5 -7.92 20 -37 -12 420 -430 u=5
770 5 -7.92 12 -35 23 420 -430 u=5
771 0 12 -35 -20 420 -430 u=5
772 5 -7.92 36 -15 23 420 -430 u=5
773 0 36 -15 -20 420 -430 u=5
774 0 38 -23 15 420 -430 u=5
775 0 20 -37 15 420 -430 u=5
776 5 -7.92 38 -23 -12 430 -445 u=5
777 5 -7.92 20 -37 -12 430 -445 u=5
778 5 -7.92 12 -35 23 430 -445 u=5
779 0 12 -35 -20 430 -445 u=5
780 5 -7.92 36 -15 23 430 -445 u=5
781 0 36 -15 -20 430 -445 u=5
782 0 38 -23 15 430 -445 u=5
783 0 20 -37 15 430 -445 u=5
784 5 -7.92 38 -23 -12 445 -460 u=5
785 5 -7.92 20 -37 -12 445 -460 u=5
786 5 -7.92 12 -35 23 445 -460 u=5
787 0 12 -35 -20 445 -460 u=5
788 5 -7.92 36 -15 23 445 -460 u=5
789 0 36 -15 -20 445 -460 u=5
790 0 38 -23 15 445 -460 u=5
791 0 20 -37 15 445 -460 u=5
792 0 23 -12 400 -460 u=5
793 0 23 15 400 -460 u=5
794 0 15 -20 400 -460 u=5
795 0 -12 -20 400 -460 u=5
c
c egg crate
c
c storage locations
c
c 201 0 -301 -112 101 620 -675
202 0 -301 112 -113 101 620 -675

```

```

203 0 -301 113 -114 101      620 -675
c   204  0 -301 114      101      620 -675
c
205 0 -301      -111 102      620 -675
206 0 -301 111 -112 102 -101 620 -675
101 0 -301 112 -113 102 -101 620 -675
fill=3 (-13.68679 68.43395 0.0)
102 0 -301 113 -114 102 -101 620 -675
fill=2 ( 13.68679 68.43395 0.0)
207 0 -301 114 -115 102 -101 620 -675
208 0 -301 115      102      620 -675
c
c   209  0 -301      -110 103      620 -675
210 0 -301 110 -111 103 -102 620 -675
103 0      111 -112 103 -102 620 -675
fill=3 (-41.06037 41.06037 0.0)
104 0      112 -113 103 -102 620 -675
fill=1 (-13.68679 41.06037 0.0)
105 0      113 -114 103 -102 620 -675
fill=1 ( 13.68679 41.06037 0.0)
106 0      114 -115 103 -102 620 -675
fill=2 ( 41.06037 41.06037 0.0)
211 0 -301 115 -116 103 -102 620 -675
c   212  0 -301 116      103      620 -675
c
213 0 -301      -110 104 -103 620 -675
107 0 -301 110 -111 104 -103 620 -675
fill=3 (-68.43395 13.68679 0.0)
108 0      111 -112 104 -103 620 -675
fill=1 (-41.06037 13.68679 0.0)
109 0      112 -113 104 -103 620 -675
fill=1 (-13.68679 13.68679 0.0)
110 0      113 -114 104 -103 620 -675
fill=1 ( 13.68679 13.68679 0.0)
111 0      114 -115 104 -103 620 -675
fill=1 ( 41.06037 13.68679 0.0)
112 0 -301 115 -116 104 -103 620 -675
fill=2 ( 68.43395 13.68679 0.0)
214 0 -301 116      104 -103 620 -675
c
c   215  0 -301      -110 105 -104 620 -675
113 0 -301 110 -111 105 -104 620 -675
fill=4 (-68.43395 -13.68679 0.0)
114 0      111 -112 105 -104 620 -675
fill=1 (-41.06037 -13.68679 0.0)
115 0      112 -113 105 -104 620 -675
fill=1 (-13.68679 -13.68679 0.0)
116 0      113 -114 105 -104 620 -675
fill=1 ( 13.68679 -13.68679 0.0)
117 0      114 -115 105 -104 620 -675
fill=1 ( 41.06037 -13.68679 0.0)
118 0 -301 115 -116 105 -104 620 -675
fill=5 ( 68.43395 -13.68679 0.0)
216 0 -301 116      105 -104 620 -675
c
c   217  0 -301      -110      -105 620 -675
218 0 -301 110 -111 106 -105 620 -675
119 0      111 -112 106 -105 620 -675
fill=4 (-41.06037 -41.06037 0.0)
120 0      112 -113 106 -105 620 -675
fill=1 (-13.68679 -41.06037 0.0)
121 0      113 -114 106 -105 620 -675
fill=1 ( 13.68679 -41.06037 0.0)
122 0      114 -115 106 -105 620 -675
fill=5 ( 41.06037 -41.06037 0.0)

```



```

219 0 -301 115 -116 106 -105 620 -675
c 220 0 -301 116 -105 620 -675
c
221 0 -301 -111 -106 620 -675
222 0 -301 111 -112 107 -106 620 -675
123 0 -301 112 -113 107 -106 620 -675
fill=4 (-13.68679 -68.43395 0.0)
124 0 -301 113 -114 107 -106 620 -675
fill=5 ( 13.68679 -68.43395 0.0)
223 0 -301 114 -115 107 -106 620 -675
224 0 -301 115 -106 620 -675
c
c 225 0 -301 -112 -107 620 -675
226 0 -301 112 -113 -107 620 -675
227 0 -301 113 -114 -107 620 -675
c 228 0 -301 114 -107 620 -675
c
1001 5 -7.92 301 -302 610 -615 $ MPC shell
1003 5 -7.92 301 -302 615 -616 $ MPC shell
1005 5 -7.92 301 -302 616 -620 $ MPC shell
1007 5 -7.92 301 -302 620 -420 $ MPC shell
1009 5 -7.92 301 -302 420 -430 $ MPC shell
1011 5 -7.92 301 -302 430 -445 $ MPC shell
1013 5 -7.92 301 -302 445 -460 $ MPC shell
1014 5 -7.92 301 -302 460 -675 $ MPC shell
1015 5 -7.92 301 -302 675 -651 $ MPC shell
1017 5 -7.92 301 -302 651 -652 $ MPC shell
1019 5 -7.92 301 -302 652 -653 $ MPC shell
1021 5 -7.92 301 -302 653 -654 $ MPC shell
1023 5 -7.92 301 -302 654 -655 $ MPC shell
1025 5 -7.92 301 -302 655 -656 $ MPC shell
1027 5 -7.92 301 -302 656 -657 $ MPC shell
1028 5 -7.92 301 -302 657 -658 $ MPC shell
1029 5 -7.92 301 -302 658 -659 $ MPC shell
1031 5 -7.92 301 -302 659 -680 $ MPC shell
c
1051 5 -7.92 -301 610 -615 $ MPC baseplate
1052 5 -7.92 -301 615 -616 $ MPC baseplate
1053 5 -7.92 -301 616 -620 $ MPC baseplate
1060 5 -7.92 -301 675 -651 $ MPC lid
1061 5 -7.92 -301 651 -652 $ MPC lid
1062 5 -7.92 -301 652 -653 $ MPC lid
1063 5 -7.92 -301 653 -654 $ MPC lid
1064 5 -7.92 -301 654 -655 $ MPC lid
1065 5 -7.92 -301 655 -656 $ MPC lid
1066 5 -7.92 -301 656 -657 $ MPC lid
1067 5 -7.92 -301 657 -658 $ MPC lid
1068 5 -7.92 -301 658 -659 $ MPC lid
1069 5 -7.92 -301 659 -680 $ MPC lid
c
c overpack universes
c
c pedestal
c
2001 8 -7.82 -302 801 -610
2002 8 -7.82 -302 802 -801
2003 8 -7.82 -302 803 -802
2004 8 -7.82 -302 804 -803
2005 8 -7.82 -302 805 -804
c
2006 7 -2.35 -306 806 -805
2007 7 -2.35 -306 807 -806
2008 7 -2.35 -306 808 -807
2009 7 -2.35 -306 809 -808
2010 7 -2.35 -306 810 -809

```

2011	7	-2.35		-306	811	-810
2012	7	-2.35		-306	812	-811
2013	7	-2.35		-306	813	-812
2014	7	-2.35		-306	814	-813
c						
2016	8	-7.82	306	-302	806	-805
2017	8	-7.82	306	-302	807	-806
2028	8	-7.82	306	-302	808	-807
2019	8	-7.82	306	-302	809	-808
2020	8	-7.82	306	-302	810	-809
2021	8	-7.82	306	-302	811	-810
2022	8	-7.82	306	-302	812	-811
2023	8	-7.82	306	-302	813	-812
2024	8	-7.82	306	-302	814	-813
c						
overpack baseplate						
2031	8	-7.82		-302	815	-814
2032	8	-7.82		-302	816	-815
2033	7	-2.35		-302	817	-816
c						
gap between overpack and lid						
3001	9	-1.17e-3		-302	680	-901
c						
c						
lid						
c						
3002	8	-7.82		-307	901	-902
3003	8	-7.82		-307	902	-903
c						
3004	7	-2.35		-305	903	-904
3005	7	-2.35		-305	904	-905
3006	7	-2.35		-305	905	-906
3007	7	-2.35		-305	906	-907
3008	7	-2.35		-305	907	-908
3009	7	-2.35		-305	908	-909
c						
3010	8	-7.82	305	-307	903	-904
3011	8	-7.82	305	-307	904	-905
3012	8	-7.82	305	-307	905	-906
3013	8	-7.82	305	-307	906	-907
3014	8	-7.82	305	-307	907	-908
3015	8	-7.82	305	-307	908	-909
c						
3021	8	-7.82		-307	909	-910
3022	8	-7.82		-307	910	-911
3023	8	-7.82		-307	911	-912
3024	8	-7.82		-307	912	-913
c						
3030	0			-303	913	-914
3031	0			-303	914	-915
3032	0			-303	915	-916
3033	0			-303	916	-917
3034	0			-303	917	-918
c						
3035	8	-7.82	303	-304	913	-914
3036	8	-7.82	303	-304	914	-915
3037	8	-7.82	303	-304	915	-916
3038	8	-7.82	303	-304	916	-917
3039	0		303	-304	917	-918
c						
3040	7	-2.35	304	-307	913	-914
3041	7	-2.35	304	-307	914	-915
3042	7	-2.35	304	-307	915	-916
3043	7	-2.35	304	-307	916	-917
3044	0		304	-307	917	-918
c						
c						
c						
steel,concrete and air in gap between mpc and overpack						

```

c
4000 7 -2.35 302 -700 817 -816
4001 8 -7.82 302 -700 816 -815
4002 8 -7.82 302 -700 815 -814
4003 9 -1.17e-3 302 -700 814 -813
4004 9 -1.17e-3 302 -700 813 -812
4005 9 -1.17e-3 302 -700 812 -811
4006 9 -1.17e-3 302 -700 811 -810
4007 9 -1.17e-3 302 -700 810 -809
4008 9 -1.17e-3 302 -700 809 -808
4009 9 -1.17e-3 302 -700 808 -807
4010 9 -1.17e-3 302 -700 807 -806
4011 9 -1.17e-3 302 -700 806 -805
4012 9 -1.17e-3 302 -700 805 -804
4013 9 -1.17e-3 302 -700 804 -803
4014 9 -1.17e-3 302 -700 803 -802
4015 9 -1.17e-3 302 -700 802 -801
4016 9 -1.17e-3 302 -700 801 -610
4017 9 -1.17e-3 302 -700 610 -615
4018 9 -1.17e-3 302 -700 615 -620
4019 9 -1.17e-3 302 -700 620 -420
4020 9 -1.17e-3 302 -700 420 -430
4021 9 -1.17e-3 302 -700 430 -445
4022 9 -1.17e-3 302 -700 445 -460
4023 9 -1.17e-3 302 -700 460 -675
4024 9 -1.17e-3 302 -700 675 -651
4025 9 -1.17e-3 302 -700 651 -652
4026 9 -1.17e-3 302 -700 652 -653
4027 9 -1.17e-3 302 -700 653 -654
4028 9 -1.17e-3 302 -700 654 -655
4029 9 -1.17e-3 302 -700 655 -656
4030 9 -1.17e-3 302 -700 656 -657
4031 9 -1.17e-3 302 -700 657 -658
4032 9 -1.17e-3 302 -700 658 -659
4033 9 -1.17e-3 302 -700 659 -680
4034 9 -1.17e-3 302 -700 680 -901
4035 9 -1.17e-3 307 -700 901 -902
4036 9 -1.17e-3 307 -700 902 -903
4037 9 -1.17e-3 307 -700 903 -904
4038 9 -1.17e-3 307 -700 904 -905
4039 9 -1.17e-3 307 -700 905 -906
4040 9 -1.17e-3 307 -700 906 -907
4041 9 -1.17e-3 307 -700 907 -908
4042 9 -1.17e-3 307 -700 908 -909
4043 8 -7.82 307 -700 909 -910
4044 8 -7.82 307 -700 910 -911
4045 8 -7.82 307 -700 911 -912
4046 8 -7.82 307 -700 912 -913
4047 7 -2.35 307 -700 913 -914
4048 7 -2.35 307 -700 914 -915
4049 7 -2.35 307 -700 915 -916
4050 7 -2.35 307 -700 916 -917
4051 0 307 -700 917 -918

```

```

c
c importance splitting regions in overpack
c

```

```

5000 7 -2.35 700 -701 817 -816
5001 0 700 -701 816 -815 fill=10
5002 0 700 -701 815 -814 fill=10
5003 0 700 -701 814 -813 fill=11
5004 0 700 -701 813 -812 fill=11
5005 0 700 -701 812 -811 fill=11
5006 0 700 -701 811 -810 fill=11
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5120 8 -7.82 701 -311 420 -430

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6622 0 716 -717 445 -460
6623 0 716 -717 460 -675
6624 0 716 -717 675 -651
6625 0 716 -717 651 -652
6626 0 716 -717 652 -653
6627 0 716 -717 653 -654
6628 0 716 -717 654 -655
6629 0 716 -717 655 -656
6630 0 716 -717 656 -657
6631 0 716 -717 657 -658
6632 0 716 -717 658 -659
6633 0 716 -717 659 -680
6634 0 716 -717 680 -901
6635 0 716 -717 901 -902
6636 0 716 -717 902 -903
6637 0 716 -717 903 -904
6638 0 716 -717 904 -905
6639 0 716 -717 905 -906
6640 0 716 -717 906 -907
6641 0 716 -717 907 -908
6642 0 716 -717 908 -909
6643 0 716 -717 909 -910
6644 0 716 -717 910 -911
6645 0 716 -717 911 -912
6646 0 716 -717 912 -913
6647 0 716 -717 913 -914
6648 0 716 -717 914 -915
6649 0 716 -717 915 -916
6650 0 716 -717 916 -917
6651 0 716 -717 917 -918
c
c overpack universes
c overpack baseplate
10001 8 -7.82 371 -372 373 -374 u=10
10002 8 -7.82 374 -362 u=10
10003 8 -7.82 374 363 u=10
10004 9 -1.17e-3 374 362 -363 u=10
10005 8 -7.82 -371 367 -374 u=10
10006 8 -7.82 -371 373 -366 u=10
10007 9 -1.17e-3 -371 366 -367 u=10
10008 8 -7.82 -373 -362 u=10
10009 8 -7.82 -373 363 u=10
10010 9 -1.17e-3 -373 362 -363 u=10
10011 8 -7.82 372 373 -366 u=10
10012 8 -7.82 372 367 -374 u=10
10013 9 -1.17e-3 372 366 -367 u=10
c
c walls and top of bottom duct
10101 8 -7.82 361 -362 -932 u=11
10102 8 -7.82 363 -364 -932 u=11
10103 8 -7.82 362 -363 931 -932 u=11
c
10104 8 -7.82 365 -366 -932 u=11
10105 8 -7.82 367 -368 -932 u=11
10106 8 -7.82 366 -367 931 -932 u=11

```

```

c      inner and outer shell between bottom ducts
10107 8 -7.82      368 364 -932 315      u=11
10108 8 -7.82      368 -361 -932 315      u=11
10109 8 -7.82      -365 364 -932 315      u=11
10110 8 -7.82      -365 -361 -932 315      u=11
c
10111 8 -7.82      368 364 -932 -310      u=11
10112 8 -7.82      368 -361 -932 -310      u=11
10113 8 -7.82      -365 364 -932 -310      u=11
10114 8 -7.82      -365 -361 -932 -310      u=11
c      concrete and radial plates between bottom ducts
10121 8 -7.82      310 -315 391 -392 -932      u=11
10122 8 -7.82      310 -315 393 -394 -932      u=11
c
10131 7 -2.35      310 -315 394 -365 -932      u=11
10132 7 -2.35      310 -315 368 -391 -932      u=11
10133 7 -2.35      310 -315 392 364 -932      u=11
10134 7 -2.35      310 -315 -361 394 -932      u=11
10135 7 -2.35      310 -315 368 -393 -932      u=11
10136 7 -2.35      310 -315 392 -365 -932      u=11
10137 7 -2.35      310 -315 -391 -361 -932      u=11
10138 7 -2.35      310 -315 364 -393 -932      u=11
c      air and grid spacers in bottom ducts
10141 9 -1.17e-3    362 -363 -931 263 -264      u=11
10142 9 -1.17e-3    366 -367 -931 261 -262      u=11
c
10143 9 -1.17e-3    362 -201 -931 (-263:264)      u=11
10144 9 -1.17e-3    206 -363 -931 (-263:264)      u=11
10145 9 -1.17e-3    201 -202 -221 (-263:264)      u=11
10146 5 -7.92      202 -203 -221 (-273:274)      u=11
11146 9 -1.17e-3    202 -203 -221 273 -274 (-263:264)      u=11
10147 9 -1.17e-3    203 -204 -221 (-263:264)      u=11
10148 5 -7.92      204 -205 -221 (-273:274)      u=11
11148 9 -1.17e-3    204 -205 -221 273 -274 (-263:264)      u=11
10149 9 -1.17e-3    205 -206 -221 (-263:264)      u=11
10150 5 -7.92      201 -202 221 -222 (-263:264)      u=11
10151 5 -7.92      202 -203 221 -222 (-263:264)      u=11
10152 5 -7.92      203 -204 221 -222 (-263:264)      u=11
10153 5 -7.92      204 -205 221 -222 (-263:264)      u=11
10154 5 -7.92      205 -206 221 -222 (-263:264)      u=11
10155 9 -1.17e-3    201 -202 222 -223 (-263:264)      u=11
10156 5 -7.92      202 -203 222 -223 (-263:264)      u=11
10157 9 -1.17e-3    203 -204 222 -223 (-263:264)      u=11
10158 5 -7.92      204 -205 222 -223 (-263:264)      u=11
10159 9 -1.17e-3    205 -206 222 -223 (-263:264)      u=11
10160 5 -7.92      201 -202 223 -224 (-263:264)      u=11
10161 5 -7.92      202 -203 223 -224 (-263:264)      u=11
10162 5 -7.92      203 -204 223 -224 (-263:264)      u=11
10163 5 -7.92      204 -205 223 -224 (-263:264)      u=11
10164 5 -7.92      205 -206 223 -224 (-263:264)      u=11
10165 9 -1.17e-3    201 -202 224 -225 (-263:264)      u=11
10166 5 -7.92      202 -203 224 -225 (-263:264)      u=11
10167 9 -1.17e-3    203 -204 224 -225 (-263:264)      u=11
10168 5 -7.92      204 -205 224 -225 (-263:264)      u=11
10169 9 -1.17e-3    205 -206 224 -225 (-263:264)      u=11
10170 9 -1.17e-3    201 -206 225 -931 (-263:264)      u=11
c
10243 9 -1.17e-3    366 -211 -931 (-261:262)      u=11
10244 9 -1.17e-3    216 -367 -931 (-261:262)      u=11
10245 9 -1.17e-3    211 -212 -221 (-261:262)      u=11
10246 5 -7.92      212 -213 -221 (-261:262)      u=11
10247 9 -1.17e-3    213 -214 -221 (-261:262)      u=11
10248 5 -7.92      214 -215 -221 (-261:262)      u=11
10249 9 -1.17e-3    215 -216 -221 (-261:262)      u=11
10250 5 -7.92      211 -212 221 -222 (-261:262)      u=11

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```

10251  5 -7.92      212 -213 221 -222 (-261:262) u=11
10252  5 -7.92      213 -214 221 -222 (-261:262) u=11
10253  5 -7.92      214 -215 221 -222 (-261:262) u=11
10254  5 -7.92      215 -216 221 -222 (-261:262) u=11
10255  9 -1.17e-3    211 -212 222 -223 (-261:262) u=11
10256  5 -7.92      212 -213 222 -223 (-261:262) u=11
10257  9 -1.17e-3    213 -214 222 -223 (-261:262) u=11
10258  5 -7.92      214 -215 222 -223 (-261:262) u=11
10259  9 -1.17e-3    215 -216 222 -223 (-261:262) u=11
10260  5 -7.92      211 -212 223 -224 (-261:262) u=11
10261  5 -7.92      212 -213 223 -224 (-261:262) u=11
10262  5 -7.92      213 -214 223 -224 (-261:262) u=11
10263  5 -7.92      214 -215 223 -224 (-261:262) u=11
10264  5 -7.92      215 -216 223 -224 (-261:262) u=11
10265  9 -1.17e-3    211 -212 224 -225 (-261:262) u=11
10266  5 -7.92      212 -213 224 -225 (-261:262) u=11
10267  9 -1.17e-3    213 -214 224 -225 (-261:262) u=11
10268  5 -7.92      214 -215 224 -225 (-261:262) u=11
10269  9 -1.17e-3    215 -216 224 -225 (-261:262) u=11
10270  9 -1.17e-3    211 -216 225 -931 (-261:262) u=11
c
c      inner, outer shells and concrete between top and bot ducts
c
10301  8 -7.82      932      -311 u=11
10302  8 -7.82      932      315 u=11
10303  8 -7.82  932      311 -315 391 -392 u=11
10304  8 -7.82  932      311 -315 393 -394 u=11
10305  7 -2.35  932      311 -315 394 -391 u=11
10306  7 -2.35  932      311 -315 392 394 u=11
10307  7 -2.35  932      311 -315 392 -393 u=11
10308  7 -2.35  932      311 -315 -391 -393 u=11
c
11302  8 -7.82      315 u=12
11303  8 -7.82      -315 391 -392 u=12
11304  8 -7.82      -315 393 -394 u=12
11305  7 -2.35      -315 394 -391 u=12
11306  7 -2.35      -315 392 394 u=12
11307  7 -2.35      -315 392 -393 u=12
11308  7 -2.35      -315 -391 -393 u=12
c
13303  8 -7.82      391 -392 u=112
13304  8 -7.82      393 -394 u=112
13305  7 -2.35      394 -391 u=112
13306  7 -2.35      392 394 u=112
13307  7 -2.35      392 -393 u=112
13308  7 -2.35      -391 -393 u=112
c
12301  8 -7.82      -933 -311 u=13
12302  8 -7.82      -933 315 u=13
12303  8 -7.82      -933 311 -315 391 -392 u=13
12304  8 -7.82      -933 311 -315 393 -394 u=13
12305  7 -2.35      -933 311 -315 394 -391 u=13
12306  7 -2.35      -933 311 -315 392 394 u=13
12307  7 -2.35      -933 311 -315 392 -393 u=13
12308  7 -2.35      -933 311 -315 -391 -393 u=13
c      top duct bottom plates
10309  8 -7.82  933 -934 351 -354 u=13
10310  8 -7.82  933 -934 355 -358 u=13
c      top duct walls
10311  8 -7.82  934      351 -352 u=13
10312  8 -7.82  934      353 -354 u=13
10313  8 -7.82  934      355 -356 u=13
10314  8 -7.82  934      357 -358 u=13
c      inner and outer shell between top ducts
10407  8 -7.82      358 354 933 315 u=13

```

```

10408 8 -7.82      358 -351  933 315  u=13
10409 8 -7.82     -355 354  933 315  u=13
10410 8 -7.82     -355 -351  933 315  u=13
c
10411 8 -7.82      358 354  933 -310 u=13
10412 8 -7.82      358 -351  933 -310 u=13
10413 8 -7.82     -355 354  933 -310 u=13
10414 8 -7.82     -355 -351  933 -310 u=13
c  concrete and radial plates next to top ducts
10421 8 -7.82      310 -315 391 -392  933 -935 u=13
10422 8 -7.82      310 -315 393 -394  933 -935 u=13
c
10431 7 -2.35      310 -315 394 -355  933 -935 u=13
10432 7 -2.35      310 -315 358 -391  933 -935 u=13
10433 7 -2.35      310 -315 392  354  933 -935 u=13
10434 7 -2.35      310 -315 -351 394  933 -935 u=13
10435 7 -2.35      310 -315 358 -393  933 -935 u=13
10436 7 -2.35      310 -315 392 -355  933 -935 u=13
10437 7 -2.35      310 -315 -391 -351  933 -935 u=13
10438 7 -2.35      310 -315 354 -393  933 -935 u=13
c
c  air and grid spacers in top ducts
10441 9 -1.17e-3   352 -353  934 263 -264 u=13
10442 9 -1.17e-3   356 -357  934 261 -262 u=13
c
10443 9 -1.17e-3   352 -231  934 (-263:264) u=13
10444 9 -1.17e-3   236 -353  934 (-263:264) u=13
c
10445 9 -1.17e-3   231 -232  934 -251 (-263:264) u=13
10446 5 -7.92      232 -233  934 -251 (-263:264) u=13
10447 9 -1.17e-3   233 -234  934 -251 (-263:264) u=13
10448 5 -7.92      234 -235  934 -251 (-263:264) u=13
10449 9 -1.17e-3   235 -236  934 -251 (-263:264) u=13
10450 5 -7.92      231 -232  251 -252 (-263:264) u=13
10451 5 -7.92      232 -233  251 -252 (-263:264) u=13
10452 5 -7.92      233 -234  251 -252 (-263:264) u=13
10453 5 -7.92      234 -235  251 -252 (-263:264) u=13
10454 5 -7.92      235 -236  251 -252 (-263:264) u=13
10455 9 -1.17e-3   231 -232  252 -253 (-263:264) u=13
10456 5 -7.92      232 -233  252 -253 (-263:264) u=13
10457 9 -1.17e-3   233 -234  252 -253 (-263:264) u=13
10458 5 -7.92      234 -235  252 -253 (-263:264) u=13
10459 9 -1.17e-3   235 -236  252 -253 (-263:264) u=13
10470 9 -1.17e-3   231 -236  253      (-263:264) u=13
c
10543 9 -1.17e-3   356 -241  934 (-261:262) u=13
10544 9 -1.17e-3   246 -357  934 (-261:262) u=13
c
10545 9 -1.17e-3   241 -242  934 -251 (-261:262) u=13
10546 5 -7.92      242 -243  934 -251 (-261:262) u=13
10547 9 -1.17e-3   243 -244  934 -251 (-261:262) u=13
10548 5 -7.92      244 -245  934 -251 (-261:262) u=13
10549 9 -1.17e-3   245 -246  934 -251 (-261:262) u=13
10550 5 -7.92      241 -242  251 -252 (-261:262) u=13
10551 5 -7.92      242 -243  251 -252 (-261:262) u=13
10552 5 -7.92      243 -244  251 -252 (-261:262) u=13
10553 5 -7.92      244 -245  251 -252 (-261:262) u=13
10554 5 -7.92      245 -246  251 -252 (-261:262) u=13
10555 9 -1.17e-3   241 -242  252 -253 (-261:262) u=13
10556 5 -7.92      242 -243  252 -253 (-261:262) u=13
10557 9 -1.17e-3   243 -244  252 -253 (-261:262) u=13
10558 5 -7.92      244 -245  252 -253 (-261:262) u=13
10559 9 -1.17e-3   245 -246  252 -253 (-261:262) u=13
10570 9 -1.17e-3   241 -246  253      (-261:262) u=13
c  top plate

```

```

10641  8 -7.82      358 354  935 310 -315  u=13
10642  8 -7.82      358 -351 935 310 -315  u=13
10643  8 -7.82     -355 354  935 310 -315  u=13
10644  8 -7.82     -355 -351 935 310 -315  u=13
c
10701  8 -7.82     -314 u=15
10702  0          314 u=15
c
10711  7 -2.35      -312      u=14
10712  8 -7.82      312 -313 u=14
10713  0          313      u=14
c
99999  0 -817:918:717:(716 -816)
c
c      BLANK LINE

c      BLANK LINE
c
c      MPC surfaces\ / \ / \ / \ / \ /
c
10      px          -12.169775
11      px          -12.017375
12      px          -11.826875
13      px          -11.1125
14      px          11.1125
15      px          11.826875
16      px          12.017375
17      px          12.169775
18      py          -12.169775
19      py          -12.017375
20      py          -11.826875
21      py          -11.1125
22      py          11.1125
23      py          11.826875
24      py          12.017375
25      py          12.169775
c
26      py          -9.525
27      py          9.525
28      px          -9.525
29      px          9.525
30      py          -6.35
31      py          6.35
32      px          -6.35
33      px          6.35
c
35      px          -11.46969
36      px          11.46969
37      py          -11.46969
38      py          11.46969
c
40      px          -10.8204
41      px          10.8204
42      py          -10.8204
43      py          10.8204
c
101     py          82.12074
102     py          54.74716
103     py          27.37358
104     py          0.0
105     py          -27.37358
106     py          -54.74716
107     py          -82.12074
c
116     px          82.12074

```



```

c    bottom duct planes
361 px    -20.955  $ start of item 13
362 px    -19.05  $ end of item 13
363 px     19.05  $ start of item 13
364 px     20.955  $ end of item 13
c
365 py    -20.955  $ start of item 13
366 py    -19.05  $ end of item 13
367 py     19.05  $ start of item 13
368 py     20.955  $ end of item 13
c    cutouts in item 1
371 px    -123.19
372 px     123.19
373 py    -123.19
374 py     123.19
c    item 14
391 1 py    -0.9525  $ steel plate in concrete at 45/225 degrees
392 1 py     0.9525  $ steel plate in concrete at 45/225 degrees
393 1 px    -0.9525  $ steel plate in concrete at 135/315 degrees
394 1 px     0.9525  $ steel plate in concrete at 135/315 degrees
c
c    bottom shielding cross plates
c
201 px    -18.57375
202 px     -6.35
203 px    -5.715
204 px     5.715
205 px     6.35
206 px    18.57375
c
211 py    -18.57375
212 py     -6.35
213 py    -5.715
214 py     5.715
215 py     6.35
216 py    18.57375
c
221 pz    -32.8168
222 pz    -32.1818
223 pz    -24.3586
224 pz    -23.7236
225 pz    -15.9004
c
c    top shielding cross plates
c
231 px    -31.27375
232 px    -10.795
233 px    -10.16
234 px     10.16
235 px     10.795
236 px    31.27375
c
241 py    -31.27375
242 py    -10.795
243 py    -10.16
244 py     10.16
245 py     10.795
246 py    31.27375
c
251 pz     523.24
252 pz     523.875
253 pz     530.86
c    end of cross plates in openings
261 px    -107.315
262 px     107.315

```

263 py -107.315  
 264 py 107.315  
 c end of part of bottom cross plates  
 271 px -124.46  
 272 px 124.46  
 273 py -124.46  
 274 py 124.46  
 c  
 c radial planes in overpack  
 700 cz 93.345 \$ ID of overpack  
 701 cz 95.885  
 702 cz 98.5 \$ slightly diff from 311  
 703 cz 103.505  
 704 cz 108.585  
 705 cz 113.665  
 706 cz 118.745  
 707 cz 123.825  
 708 cz 128.905  
 709 cz 133.985  
 710 cz 139.065  
 711 cz 144.145  
 712 cz 149.225  
 713 cz 154.305  
 714 cz 159.385  
 715 cz 164.465  
 716 cz 168.275  
 717 cz 169.275  
 c  
 c planes in pedestal  
 c  
 801 pz 12.7  
 802 pz 10.16  
 803 pz 7.62  
 804 pz 5.08  
 805 pz 2.54 \$ bottom of item 24  
 806 pz -2.54  
 807 pz -7.62  
 808 pz -12.7  
 809 pz -17.78  
 810 pz -22.86  
 811 pz -27.94  
 812 pz -33.02  
 813 pz -38.1  
 814 pz -40.64 \$ start of item 1  
 815 pz -43.18 \$  
 816 pz -45.72 \$ ground  
 817 pz -76.20  
 c  
 c planes in lid  
 c  
 901 pz 501.65 \$ start of item 6  
 902 pz 502.285 \$ 0.25 inch segment from start  
 903 pz 504.825 \$ end of item 6  
 904 pz 509.905  
 905 pz 513.715  
 906 pz 516.3 \$ end of item 8 plus a little  
 907 pz 521.335  
 908 pz 526.415  
 909 pz 531.495 \$ end of concrete start of item 10  
 910 pz 534.035  
 911 pz 536.575  
 912 pz 539.115  
 913 pz 541.655 \$ end of item 10  
 914 pz 546.735  
 915 pz 551.815



916	pz	556.895	
917	pz	561.975	
918	pz	562.975	
c			
c	planes in overpack		
c			
931	pz	-15.24	\$ bottom of item 11
932	pz	-10.16	\$ top of item 11
933	pz	513.08	\$ bottom of item 8 and top of item 28
934	pz	516.255	\$ top of item 8
935	pz	529.59	\$ start of item 9
c			
c	for tallying		
c			
501	pz	-45.72	
502	pz	-30.48	
503	pz	-15.24	
504	pz	0.00	
505	pz	15.24	
506	pz	30.48	
507	pz	45.72	
508	pz	60.96	
509	pz	76.20	
510	pz	91.44	
511	pz	106.68	
512	pz	121.92	
513	pz	137.16	
514	pz	152.40	
515	pz	167.64	
516	pz	182.88	
517	pz	198.12	
518	pz	213.36	
519	pz	228.60	
520	pz	243.84	
521	pz	259.08	
522	pz	274.32	
523	pz	289.56	
524	pz	304.80	
525	pz	320.04	
526	pz	335.28	
527	pz	350.52	
528	pz	365.76	
529	pz	381.00	
530	pz	396.24	
531	pz	411.48	
532	pz	426.72	
533	pz	441.96	
534	pz	457.20	
535	pz	472.44	
536	pz	487.68	
537	pz	502.92	
538	pz	518.16	
539	pz	533.40	
c			
550	cz	15.24	
551	cz	30.48	
552	cz	45.72	
553	cz	60.96	
554	cz	76.20	
555	cz	91.44	
556	cz	106.68	
557	cz	121.92	
558	cz	137.16	
559	cz	152.40	
560	cz	167.64	

```

c
c   BLANK LINE

c   BLANK LINE
c
*tr1   0 0 0 45   315   90 135   45   90 90 90 0
c
c       PHOTON MATERIALS
c
c   fuel 3.4 w/o U235   10.412 gm/cc
m1      92235.01p   -0.029971
        92238.01p   -0.851529
        8016.01p    -0.1185
c   homogenized fuel density 3.8699 gm/cc
m2      92235.01p   -0.027652
        92238.01p   -0.719715
        8016.01p    -0.100469
        40000.01p   -0.149015
        50000.01p   -0.002587
        26000.01p   -0.000365
        24000.01p   -0.000198
c   zirconium 6.55 gm/cc
m3      40000.01p   1.           $ Zr Clad
c   stainless steel 7.92 gm/cc
m5      24000.01p   -0.19
        25055.01p   -0.02
        26000.01p   -0.695
        28000.01p   -0.095
c   boral 2.644 gm/cc
m6      5010.01p    -0.044226
        5011.01p    -0.201474
        13027.01p   -0.6861
        6000.01p    -0.0682
c   Concrete (NBS Ordinary) @ 2.35 g/cc (Ref: LA-12827-M)
m7      14000.01p   -0.315
        13027.01p   -0.048
        8016.01p    -0.500
        1001.01p    -0.006
        11023.01p   -0.017
        20000.01p   -0.083
        26000.01p   -0.012
        19000.01p   -0.019
c   carbon steel 7.82 gm/cc
m8      6000.01p   -0.005 26000.01p -0.995
c   air density 1.17e-3 gm/cc
m9      7014.01p 0.78 8016.01p 0.22
c
c       NEUTRON MATERIALS
c
c   fuel 3.4 w/o U235   10.412 gm/cc
c   m1      92235.50c   -0.029971
        92238.50c   -0.851529
        8016.50c    -0.1185
c   c   homogenized fuel density 3.8699 gm/cc
c   m2      92235.50c   -0.027652
        92238.50c   -0.719715
        8016.50c    -0.100469
        40000.35c   -0.149015
        50000.35c   -0.002587
        26000.55c   -0.000365
        24000.50c   -0.000198
c   c   helium 1e-4 gm/cc
c   m3      2004.50c 1.0
c   c   stainless steel 7.92 gm/cc
c   m5      24000.50c   -0.19

```

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c          25055.50c  -0.02
c          26000.55c  -0.695
c          28000.50c  -0.095
c  c      boral  2.644 gm/cc
c  m6          5010.50c  -0.044226
c          5011.56c  -0.201474
c          13027.50c  -0.6861
c          6000.50c  -0.0682
c  c      Concrete (NBS Ordinary) @ 2.35 g/cc (Ref: LA-12827-M)
c  m7          14000.50c  -0.315
c          13027.50c  -0.048
c          8016.50c  -0.500
c          1001.50c  -0.006
c          11023.50c  -0.017
c          20000.50c  -0.083
c          26000.55c  -0.012
c          19000.50c  -0.019
c  mt7      lwtr.01t
c  c      carbon steel 7.82 gm/cc
c  m8      6000.50c -0.005 26000.55c -0.995
c  c      air density 1.17e-3 gm/cc
c  m9      7014.50c 0.78 8016.50c 0.22
c
phys:n  20 0.0
phys:p  100 0
c  imp:n  1 228r 0
c  imp:p  1 228r 0
nps      13500000
prdmp   j  -60  1  2
c  print  10 110 160 161 20 170
print
mode p
ssw     716 917
c
sdef  par=2  erg=d1  axs=0 0 1  x=d4  y=fx  d5  z=d3
c
c  energy dist for gammas in the fuel
c
c  si1  h  0.7 1.0 1.5 2.0 2.5 3.0
c  sp1   0  0.43 0.27 0.22 0.04 0.04
c
c  energy dist for neutrons in the fuel
c
c  si1  h  0.1 0.4 0.9 1.4 1.85 3.0 6.43 20.0
c  sp1   0  0.03787 0.1935 0.1773 0.1310 0.2320 0.2098 0.01853
c
c  energy dist for Co60 gammas
c
si1  l  1.3325 1.1732
sp1   0.5  0.5
c
c  axial dist for phot in fuel
c
c  si3  h  53.2765 68.5165 83.7565 114.2365 175.1965 236.1565
c          297.1165 358.0765 388.5565 403.7965 419.0365
c  sp3  0  0.022854 0.035321 0.08975 0.184167 0.183 0.179833
c          0.175017 0.080033 0.030575 0.019458
c  sb3  0  1 1 1 1 1 1 1 1 1
c
c  axial dist for Co60 - a zero prob is in the fuel
c
si3  h  32.004 50.7365 419.0365 428.72025 439.83275 452.6915
sp3  0  0.44 0.0 0.05 0.05 0.46
sb3  0  0.50 0.0 0.05 0.10 0.35
c

```

```

si4  s          13 14
          12 13 14 15
        11 12 13 14 15 16
        11 12 13 14 15 16
          12 13 14 15
          13 14

sp4  1 23r
c
ds5  s          26 26
          25 25 25 25
          24 24 24 24 24 24
          23 23 23 23 23 23
          22 22 22 22
          21 21

c
si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639
si14  2.86639  24.50719
si15  30.23997  51.88077
si16  57.61355  79.25435
c
si21 -79.25435 -57.61355
si22 -51.88077 -30.23997
si23 -24.50719 -2.86639
si24  2.86639  24.50719
si25  30.23997  51.88077
si26  57.61355  79.25435
c
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
c
#  imp:p
314  2
315  2
316  2
317  2
318  2
319  2
320  2
321  2
346  2
353  2
360  2
361  2
362  2
363  2
364  2
365  2
366  2
367  2
414  2
415  2
416  2
417  2

```

418 2  
419 2  
420 2  
421 2  
446 2  
453 2  
460 2  
461 2  
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463 2  
464 2  
465 2  
466 2  
467 2  
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753 2  
760 2  
761 2  
762 2  
763 2  
764 2

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766	2
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791	4
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651	8
751	8
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1052	8
1053	4
1060	24
1061	24
1062	72
1063	72
1064	216
1065	216
1066	648
1067	648
1068	1944
1069	1944
1001	16
1003	8
1005	8
1007	4

1009	2
1011	4
1013	8
1014	16
1015	48
1017	24
1019	72
1021	72
1023	216
1025	216
1027	648
1028	648
1029	1944
1031	1944
c	
2001	32
2002	64
2003	128
2004	256
2005	512
2006	1024
2007	1024
2008	2048
2009	2048
2010	4096
2011	4096
2012	8192
2013	8192
2014	16384
c	
2016	1024
2017	1024
2028	2048
2019	2048
2020	4096
2021	4096
2022	8192
2023	8192
2024	16384
c	
2031	16384
2032	16384
2033	16384
c	
3001	3888
3002	3888
3003	3888
c	
3004	7776
3005	7776
3006	7776
3007	7776
3008	15552
3009	15552
c	
3010	7776
3011	7776
3012	7776
3013	7776
3014	15552
3015	15552
c	
3021	46656
3022	46656
3023	139968

3024	139968
c	
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3031	1
3032	1
3033	1
3034	1
c	
3035	279936
3036	279936
3037	559872
3038	559872
3039	1
c	
3040	279936
3041	279936
3042	559872
3043	559872
3044	1
c	
4000	32768
4001	32768
4002	32768
4003	32768
4004	16384
4005	16384
4006	8192
4007	8192
4008	4096
4009	4096
4010	2048
4011	2048
4012	1024
4013	512
4014	256
4015	128
4016	64
4017	32
4018	16
4019	8
4020	4
4021	8
4022	16
4023	32
4024	96
4025	48
4026	144
4027	144
4028	432
4029	432
4030	1296
4031	1296
4032	3888
4033	3888
4034	7776
4035	7776
4036	7776
4037	15552
4038	15552
4039	15552
4040	15552
4041	31104
4042	31104
4043	93312
4044	93312

4045	279936
4046	279936
4047	559872
4048	559872
4049	1119744
4050	1119744
4051	1

c

5000	32768
5001	32768
5002	32768
5003	32768
5004	16384
5005	16384
5006	8192
5007	8192
5008	4096
5009	4096
5010	2048
5011	2048
5012	1024
5013	512
5014	256
5015	128
5016	64
5017	32
5018	16
5019	8
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5021	8
5022	16
5023	32
5024	96
5025	48
5026	144
5027	144
5028	432
5029	432
5030	1296
5031	1296
5032	3888
5033	3888
5034	7776
5035	7776
5036	7776
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5039	15552
5040	15552
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5043	93312
5044	93312
5045	279936
5046	279936
5047	559872
5048	559872
5049	1119744
5050	1119744
5051	1

c

5100	65536
5101	65536
5102	65536
5103	65536

5104	32768
5105	32768
5106	16384
5107	16384
5108	8192
5109	8192
5110	4096
5111	4096
5112	2048
5113	1024
5114	512
5115	256
5116	128
5117	64
5118	32
5119	16
5120	8
5121	16
5122	32
5123	64
5124	192
5125	96
5126	288
5127	288
5128	864
5129	864
5130	2592
5131	2592
5132	7776
5133	7776
5134	15552
5135	15552
5136	15552
5137	31104
5138	31104
5139	31104
5140	31104
5141	62208
5142	62208
5143	186624
5144	186624
5145	559872
5146	559872
5147	1119744
5148	1119744
5149	2239488
5150	2239488
5151	1
c	
5200	131072
5201	131072
5202	131072
5203	131072
5204	65536
5205	65536
5206	32768
5207	32768
5208	16384
5209	16384
5210	8192
5211	8192
5212	4096
5213	2048
5214	1024
5215	512

5216	256
5217	128
5218	64
5219	32
5220	16
5221	32
5222	64
5223	128
5224	384
5225	192
5226	576
5227	576
5228	1728
5229	1728
5230	5184
5231	5184
5232	15552
5233	15552
5234	31104
5235	31104
5236	31104
5237	62208
5238	62208
5239	62208
5240	62208
5241	124416
5242	124416
5243	373248
5244	373248
5245	1119744
5246	1119744
5247	2239488
5248	2239488
5249	4478976
5250	4478976
5251	1
c	
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5301	262144
5302	262144
5303	262144
5304	131072
5305	131072
5306	65536
5307	65536
5308	32768
5309	32768
5310	16384
5311	16384
5312	8192
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5314	2048
5315	1024
5316	512
5317	256
5318	128
5319	64
5320	32
5321	64
5322	128
5323	256
5324	768
5325	384
5326	1152
5327	1152

5328	3456
5329	3456
5330	10368
5331	10368
5332	31104
5333	31104
5334	62208
5335	62208
5336	62208
5337	124416
5338	124416
5339	124416
5340	124416
5341	248832
5342	248832
5343	746496
5344	746496
5345	2239488
5346	2239488
5347	4478976
5348	4478976
5349	8957952
5350	8957952
5351	1
c	
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5501	524288
5502	524288
5503	524288
5504	262144
5505	262144
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5507	131072
5508	65536
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5514	4096
5515	2048
5516	1024
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5518	256
5519	128
5520	64
5521	128
5522	256
5523	512
5524	1536
5525	768
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5527	2304
5528	6912
5529	6912
5530	20736
5531	20736
5532	62208
5533	62208
5534	124416
5535	124416
5536	124416
5537	248832
5538	248832
5539	248832

5540	248832
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5542	497664
5543	1492992
5544	1492992
5545	4478976
5546	4478976
5547	1
5548	1
5549	1
5550	1
5551	1

c

5700	1048576
5701	1048576
5702	1048576
5703	1048576
5704	524288
5705	524288
5706	262144
5707	262144
5708	131072
5709	131072
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5711	65536
5712	32768
5713	16384
5714	8192
5715	4096
5716	2048
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5719	256
5720	128
5721	256
5722	512
5723	1024
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5727	4608
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5729	13824
5730	41472
5731	41472
5732	124416
5733	124416
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5736	248832
5737	497664
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5739	497664
5740	497664
5741	995328
5742	995328
5743	2985984
5744	2985984
5745	8957952
5746	8957952

c

5900	2097152
5901	2097152
5902	2097152
5903	2097152



5904	1048576
5905	1048576
5906	524288
5907	524288
5908	262144
5909	262144
5910	131072
5911	131072
5912	65536
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5914	16384
5915	8192
5916	4096
5917	2048
5918	1024
5919	512
5920	256
5921	512
5922	1024
5923	2048
5924	6144
5925	3072
5926	9216
5927	9216
5928	27648
5929	27648
5930	82944
5931	82944
5932	248832
5933	248832
5934	497664
5935	497664
5936	497664
5937	995328
5938	995328
5939	995328
5940	995328
5941	1990656
5942	1990656
5943	5971968
5944	5971968
5945	17915904
5946	17915904
c	
6100	4194304
6101	4194304
6102	4194304
6103	4194304
6104	2097152
6105	2097152
6106	1048576
6107	1048576
6108	524288
6109	524288
6110	262144
6111	262144
6112	131072
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6114	32768
6115	16384
6116	8192
6117	4096
6118	2048
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6120	512

6121	1024
6122	2048
6123	4096
6124	12288
6125	6144
6126	18432
6127	18432
6128	55296
6129	55296
6130	165888
6131	165888
6132	497664
6133	497664
6134	995328
6135	995328
6136	995328
6137	1990656
6138	1990656
6139	1990656
6140	1990656
6141	3981312
6142	3981312
6143	11943936
6144	11943936
6145	35831808
6146	35831808
c	
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6301	8388608
6302	8388608
6303	8388608
6304	4194304
6305	4194304
6306	2097152
6307	2097152
6308	1048576
6309	1048576
6310	524288
6311	524288
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6313	131072
6314	65536
6315	32768
6316	16384
6317	8192
6318	4096
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6320	1024
6321	2048
6322	4096
6323	8192
6324	24576
6325	12288
6326	36864
6327	36864
6328	110592
6329	110592
6330	331776
6331	331776
6332	995328
6333	995328
6334	1990656
6335	1990656
6336	1990656
6337	3981312

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6339	3981312
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6341	7962624
6342	7962624
6343	23887872
6344	23887872
6345	71663616
6346	71663616
c	
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6401	8388608
6402	8388608
6403	8388608
6404	4194304
6405	4194304
6406	2097152
6407	2097152
6408	1048576
6409	1048576
6410	524288
6411	524288
6412	262144
6413	131072
6414	65536
6415	32768
6416	16384
6417	8192
6418	4096
6419	2048
6420	1024
6421	2048
6422	4096
6423	8192
6424	24576
6425	12288
6426	36864
6427	36864
6428	110592
6429	110592
6430	331776
6431	331776
6432	995328
6433	995328
6434	1990656
6435	1990656
6436	1990656
6437	3981312
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6439	3981312
6440	3981312
6441	7962624
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6443	23887872
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6445	71663616
6446	71663616
c	
6500	16777216
6501	16777216
6502	16777216
6503	16777216
6504	8388608
6505	8388608
6506	4194304

6507	4194304
6508	2097152
6509	2097152
6510	1048576
6511	1048576
6512	524288
6513	262144
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c      2.5e-8  1.0e-7  1.0e-6  1.0e-5  1.0e-4  1.0e-3  1.0e-2  0.1
c      0.5    1.0    2.5    5.0    7.0    10.0   14.0   20.0
c      3.67e-6 3.67e-6 4.46e-6 4.54e-6 4.18e-6 3.76e-6 3.56e-6 2.17e-5
c      9.26e-5 1.32e-4 1.25e-4 1.56e-4 1.47e-4 1.47e-4 2.08e-4 2.27e-4
c
c      photon dose factors
c
c      0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
c      0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
c      3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
c      13.0  15.0
c      3.96e-06 5.82e-07 2.90e-07 2.58e-07 2.83e-07 3.79e-07 5.01e-07
c      6.31e-07 7.59e-07 8.78e-07 9.85e-07 1.08e-06 1.17e-06 1.27e-06
c      1.36e-06 1.44e-06 1.52e-06 1.68e-06 1.98e-06 2.51e-06 2.99e-06
c      3.42e-06 3.82e-06 4.01e-06 4.41e-06 4.83e-06 5.23e-06 5.60e-06
c      5.80e-06 6.01e-06 6.37e-06 6.74e-06 7.11e-06 7.66e-06 8.77e-06
c      1.03e-05 1.18e-05 1.33e-05
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c      PHOTON TALLIES
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c      f102:p 716 917
c      ft102  scx 3
c      de102  0.01 0.03 0.05 0.07 0.1 0.15 0.2 0.25 0.3 0.35 0.4 0.45
c      0.5 0.55 0.6 0.65 0.7 0.8 1.0 1.4 1.8 2.2 2.6 2.8 3.25
c      3.75 4.25 4.75 5.0 5.25 5.75 6.25 6.75 7.5 9.0 11.0
c      13.0 15.0
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c      6.31e-07 7.59e-07 8.78e-07 9.85e-07 1.08e-06 1.17e-06 1.27e-06
c      1.36e-06 1.44e-06 1.52e-06 1.68e-06 1.98e-06 2.51e-06 2.99e-06
c      3.42e-06 3.82e-06 4.01e-06 4.41e-06 4.83e-06 5.23e-06 5.60e-06
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c      1.03e-05 1.18e-05 1.33e-05
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**APPENDIX 5.D**

**DOSE RATE COMPARISON FOR DIFFERENT COBALT IMPURITY LEVELS**

**(Total number of pages in this appendix : 6)**

The dose rate adjacent to and one meter from the 100-ton HI-TRAC and the HI-STORM overpack are presented on Tables 5.D.1 through 5.D.4 for the MPC-24 with different burnup and cooling times and different assumed Cobalt-59 impurity levels for inconel. The HI-TRAC results were calculated for an earlier design which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below. The following burnup and cooling time combinations are presented.

100-ton HI-TRAC

- 35,000 MWD/MTU and 5 year cooling  
1000 ppm (1.0 gm/kg) Cobalt-59 impurity in inconel
- 45,000 MWD/MTU and 9 year cooling  
4700 ppm (4.7 gm/kg) Cobalt-59 impurity in inconel

HI-STORM

- 45,000 MWD/MTU and 5 year cooling  
1000 ppm (1.0 gm/kg) Cobalt-59 impurity in inconel
- 45,000 MWD/MTU and 9 year cooling  
4700 ppm (4.7 gm/kg) Cobalt-59 impurity in inconel

On Tables 5.D.1 through 5.D.4, the contribution to the dose rate from activation in incore grid spacers is explicitly shown.

These results demonstrate that the dose rates at the longer cooling time are essentially equivalent to (within 11%) or bounded by the dose rates at the shorter cooling times even though a very conservative Cobalt-59 impurity level of 4700 ppm was assumed for the longer cooling times.

Table 5.2.1 shows the masses of inconel and steel that are used in the modeling of the PWR fuel assembly. When 4700 ppm was used for the impurity level in the inconel, an effective Cobalt-59 impurity level was used for the regions containing both steel and inconel. The following table summarizes the impurity levels that were used.

Region	Regional Co-59 impurity when 1000 ppm in inconel assumed	Regional Co-59 impurity when 4700 ppm in inconel assumed
Lower end fitting	1000 ppm	1340 ppm
Incore grid spacers	1000 ppm	4700 ppm
Gas plenum springs	1000 ppm	3417 ppm
Gas plenum spacer	1000 ppm	3417 ppm
Upper end fitting	1000 ppm	1000 ppm

Table 5.D.1

DOSE RATES ADJACENT TO 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point <sup>†</sup> Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
45,000 MWD/MTU AND 9-YEAR COOLING						
1	12.20	11.89	12.65	507.16	171.98	715.87
2	345.15	332.90	52.81	0.75	88.26	819.87
3	4.72	5.19	1.98	369.24	214.70	595.83
4	7.13	8.11	0.97	197.76	180.55	394.52
5 (pool lid)	48.63	54.68	17.52	2557.68	1194.17	3872.68
5 (transfer lid)	137.52	155.17	0.97	3811.45	674.42	4779.53
1000 ppm Co-59 in inconel						
35,000 MWD/MTU AND 5-YEAR COOLING						
1	3.73	27.05	6.51	543.06	88.57	668.92
2	105.37	696.56	27.19	0.80	45.46	875.38
3	1.44	11.44	1.02	473.51	110.57	597.98
4	2.18	17.87	0.50	241.22	92.97	354.74
5 (pool lid)	28.09	186.49	8.27	2751.19	554.51	3528.55
5 (transfer lid)	41.98	293.57	0.50	4081.28	347.33	4764.66

<sup>†</sup> Refer to Figure 5.1.4.

Table 5.D.2

DOSE RATES AT 1 METER FROM 100-TON HI-TRAC FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point <sup>†</sup> Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in inconel</b>						
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>						
1	44.32	43.03	7.14	73.99	27.98	196.46
2	148.49	143.94	16.40	6.18	32.38	347.39
3	18.76	18.25	3.88	72.75	13.83	127.47
4	2.19	2.80	0.17	61.70	44.87	111.73
5(transfer lid)	55.57	64.25	0.18	1556.99	188.18	1865.16
<b>1000 ppm Co-59 in inconel</b>						
<b>35,000 MWD/MTU AND 5-YEAR COOLING</b>						
1	13.53	91.20	3.68	79.16	14.41	201.97
2	45.33	302.99	8.44	6.13	16.68	379.57
3	5.73	38.74	2.00	71.95	7.12	125.54
4	0.67	6.21	0.09	74.47	23.11	104.55
5(transfer lid)	16.96	128.14	0.09	1667.22	96.91	1909.32

<sup>†</sup> Refer to Figure 5.1.4.

Table 5.D.3

DOSE RATES ADJACENT TO HI-STORM OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point <sup>†</sup> Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in incorel</b>					
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
1	0.54	2.95	3.86	2.37	9.72
2	7.73	10.40	0.03	1.48	19.63
3	0.36	1.97	2.59	1.18	6.11
4	0.10	0.53	0.33	3.10	4.06
<b>1000 ppm Co-59 in incorel</b>					
<b>45,000 MWD/MTU AND 5-YEAR COOLING</b>					
1	0.20	5.68	4.87	2.76	13.51
2	2.73	28.93	0.03	1.88	33.58
3	3.87	0.13	3.21	1.38	8.59
4	0.04	0.91	0.36	3.60	4.91

<sup>†</sup> Refer to Figures 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.



Table 5.D.4

DOSE RATES ONE METER FROM HI-STORM OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point <sup>†</sup> Location	Incore Grid Spacer <sup>60</sup> Co Gammas (mrem/hr)	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>4700 ppm Co-59 in incorel</b>					
<b>45,000 MWD/MTU AND 9-YEAR COOLING</b>					
1	0.77	2.01	2.30	0.46	5.54
2	3.90	5.14	0.39	0.64	10.08
3	0.44	1.09	1.72	0.18	3.42
4	0.05	0.23	0.14	0.94	1.37
<b>1000 ppm Co-59 in incorel</b>					
<b>45,000 MWD/MTU AND 5-YEAR COOLING</b>					
1	0.28	4.49	2.90	0.54	8.21
2	1.41	14.98	0.25	0.78	17.42
3	0.16	2.57	2.09	0.21	5.03
4	0.02	0.42	0.16	1.10	1.70

<sup>†</sup> Refer to Figures 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

**APPENDIX 5.E**

**Dose Rates for a HI-STORM 100 Overpack With and Without an Inner Shield Shell**

**(Total number of pages in this appendix : 4)**

In June 2001, the inner shield shell of the HI-STORM 100 overpack was removed. As a compensating change, the density of the concrete in the body of the overpack was increased to 155 lb/cuft as discussed in Section 5.3. This appendix presents a comparison of the dose rates calculated for a HI-STORM 100 overpack with and without an inner shield shell. The MPC-24 was used in this analysis. Table 5.E.1 presents the results for the overpack containing the inner shield shell and Table 5.E.2 presents the results for the overpack without the inner shield shell and the higher density concrete in the body of the overpack.

The results indicate that the change in shielding configuration does not significantly impact the dose rates. The dose rates for the surface of the ducts show a slight increase (7%) when the inner shield shell is removed while the midplane surface shows an even smaller increase (2%). The dose rates for the top of the overpack are reduced when the inner shield shell is removed and the concrete density is increased. All one meter locations are essentially identical.

Therefore, based on the results presented in this appendix, the analysis in the main body of the chapter uses the HI-STORM 100 overpack with the inner shield present.

Table 5.E.1

DOSE RATES FOR THE HI-STORM 100 OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 5-YEAR COOLING  
INNER SHIELD SHELL IS PRESENT

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Surface				
1	5.88	4.87	2.76	13.51
2	31.67	0.03	1.88	33.58
3	4.00	3.21	1.38	8.59
4	0.95	0.36	3.60	4.91
One Meter				
1	4.77	2.90	0.54	8.21
2	16.39	0.25	0.78	17.42
3	2.73	2.09	0.21	5.03
4	0.44	0.16	1.10	1.70

<sup>†</sup> Refer to Figure 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

Table 5.E.2

DOSE RATES FOR THE HI-STORM 100 OVERPACK FOR NORMAL CONDITIONS  
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 5-YEAR COOLING  
INNER SHIELD SHELL IS REMOVED

Dose Point <sup>†</sup> Location	Fuel Gammas <sup>††</sup> (mrem/hr)	<sup>60</sup> Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Surface				
1	6.48	5.67	2.28	14.43
2	32.37	0.05	1.62	34.04
3	4.23	3.67	1.24	9.14
4	0.88	0.33	3.36	4.56
One Meter				
1	4.70	3.33	0.36	8.39
2	16.70	0.30	0.69	17.69
3	2.80	1.94	0.25	4.99
4	0.40	0.18	0.94	1.51

<sup>†</sup> Refer to Figure 5.1.1.

<sup>††</sup> Gammas generated by neutron capture are included with fuel gammas.

## CHAPTER 6<sup>†</sup>: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STORM 100 System is consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor ( $k_{\text{eff}}$ ), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.
2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.
3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both.
4. Criticality safety of the cask system should not rely on use of the following credits:
  - a. burnup of the fuel
  - b. fuel-related burnable neutron absorbers
  - c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test<sup>††</sup>.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STORM 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. Analyses for the HI-STAR 100 System, which are applicable to the HI-STORM 100 System, have been previously submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

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

<sup>††</sup> *For greater credit allowance, fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.*

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor ( $k_{\text{eff}}$ ) of the HI-STORM 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STORM 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM 100 System depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24, MPC-24E and MPC-24EF);
2. The incorporation of permanent fixed neutron-absorbing panels (~~Boral~~) in the fuel basket structure;
3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and
4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24, MPC-24E and MPC-24EF, and for loading/unloading fuel in the MPC-32 and MPC-32F.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM 100 System is designed such that the fixed neutron absorber (~~Boral~~) will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the *Boral* fixed neutron absorber (~~Boral~~)
- *more than 90 percent of the B-10 content for the Metamic fixed neutron absorber, with comprehensive fabrication tests as described in Section 9.1.5.3.2.*

The following four interchangeable basket designs are available for use in the HI-STORM 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 24-cell basket (MPC-24E) for intact and damaged PWR fuel assemblies. This is a variation of the MPC-24, with an optimized cell arrangement, increased  $^{10}\text{B}$  content in the ~~Boral~~ *fixed neutron absorber* and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC). Additionally, a variation in the MPC-24E, designated MPC-24EF, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris. The MPC-24E and MPC-24EF are designed for fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 32-cell basket (MPC-32), designed for intact *and damaged* PWR fuel assemblies of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading. *Additionally, a variation in the MPC-32, designated MPC-32F, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris.*; ~~and~~
- a 68-cell basket (MPC-68), designed for both intact and damaged BWR fuel assemblies with a specified maximum planar-average enrichment. Additionally, variations in the MPC-68, designated MPC-68F and MPC-68FF, are designed for intact and damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.

*Two interchangeable neutron absorber materials are used in these baskets, Boral and Metamic. For Boral, 75 percent of the minimum B-10 content is credited in the criticality analysis, while for Metamic, 90 percent of the minimum B-10 content is credited, based on the neutron absorber*



tests specified in Section 9.1.5.3. However, the B-10 content in Metamic is chosen to be lower than the B-10 content in Boral, and is chosen so that the absolute B-10 content credited in the criticality analysis is the same for the two materials. This makes the two materials identical from a criticality perspective. This is confirmed by comparing results for a selected number of cases that were performed with both materials (see Section 6.4.11). Calculations in this chapter are therefore only performed for the Boral neutron absorber, with results directly applicable to Metamic.

The HI-STORM 100 System includes the HI-TRAC transfer cask and the HI-STORM storage cask. The HI-TRAC transfer cask is required for loading and unloading fuel into the MPC and for transfer of the MPC into the HI-STORM storage cask. HI-TRAC uses a lead shield for gamma radiation and a water-filled jacket for neutron shielding. The HI-STORM storage cask uses concrete as a shield for both gamma and neutron radiation. Both the HI-TRAC transfer cask and the HI-STORM storage cask, as well as the HI-STAR System<sup>†</sup>, accommodate the interchangeable MPC designs. The three cask designs (HI-STAR, HI-STORM, and HI-TRAC) differ only in the overpack reflector materials (steel for HI-STAR, concrete for HI-STORM, and lead for HI-TRAC), which do not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 system and vice versa. Therefore, the majority of criticality calculations to support both the HI-STAR and the HI-STORM System have been performed for only one of the two systems, namely the HI-STAR System. Only a selected number of analyses has been performed for both systems to demonstrate that this approach is valid. Therefore, unless specifically noted otherwise, all analyses documented throughout this chapter have been performed for the HI-STAR System. For the cases where analyses were performed for both the HI-STORM and HI-STAR System, this is clearly indicated.

The HI-STORM 100 System for storage (concrete overpack) is dry (no moderator), and thus, the reactivity is very low ( $k_{\text{eff}} < 0.52$ ). However, the HI-STORM 100 System for cask transfer (HI-TRAC, lead overpack) is flooded for loading and unloading operations, and thus, represents the limiting case in terms of reactivity.

The MPC-24EF, MPC-32F and MPC-68FF contains the same basket as the MPC-24E, MPC-32 and MPC-68, respectively. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF, the MPC-32 and MPC-32F, and the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-24E, MPC-32 and MPC-68 are valid for the MPC-24EF, MPC-32F and MPC-68FF, respectively, and no separate analyses for the MPC-24EF, MPC-32F and MPC-68FF are necessary. Therefore, throughout this chapter and unless otherwise noted, 'MPC-68' refers to 'MPC-68 and/or MPC-68FF', 'MPC-24E' or 'MPC-24E/EF' refers to 'MPC-24E and/or MPC-24EF', and 'MPC-32' or 'MPC-32/32F' refers to 'MPC-32 and/or MPC-32F'.

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<sup>†</sup> Analyses for the HI-STAR System have previously been submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

~~The MPC-68FF contains the same basket as the MPC-68. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-68 are valid for the MPC-68FF and no separate analyses for the MPC-68FF are necessary.~~

Confirmation of the criticality safety of the HI-STORM 100 System was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package [6.4.1]. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, and MCNP4a [6.1.4] were used. The CASMO-3 and MCNP4a calculations identify those tolerances that cause a positive reactivity effect, enabling the subsequent Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24, MPC-24E and MPC-24EF) or cell spacing (MPC-32, MPC-32F, MPC-68, MPC-68F and MPC-68FF), (3) the <sup>10</sup>B loading of the neutron absorber panels, and (4) the soluble boron concentration in the water. The critical experiment benchmarking is presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.
- 10CFR72.124, Criteria For Nuclear Criticality Safety.
- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.
- Consistent with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.
- Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber *and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber.*
- The fuel stack density is conservatively assumed to be *at least* 96% of theoretical (10.522 g/cm<sup>3</sup>) for all criticality analyses. Fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels may be slightly greater than 96% of theoretical, the actual stack density will be less.
- No credit is taken for the <sup>234</sup>U and <sup>236</sup>U in the fuel.
- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.
- When credit is taken for soluble boron, a <sup>10</sup>B content of 18.0 wt% in boron is assumed.
- Neutron absorption in minor structural members and optional heat conduction elements is neglected, i.e., spacer grids, basket supports, and optional aluminum heat conduction elements are replaced by water.
- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.
- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.

- Planar-averaged enrichments are assumed for BWR fuel. (Consistent with NUREG-1536, analysis is presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)
- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.
- For evaluation of the bias, all benchmark calculations that result in a  $k_{\text{eff}}$  greater than 1.0 are conservatively truncated to 1.0000, consistent with NUREG-1536.
- The water reflector above and below the fuel is assumed to be unborated water, 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 ~~the Certificate of Compliance~~ *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.

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. ~~To~~ *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.*, ~~results of the design basis criticality safety calculations for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System) are listed in Tables 6.1.1 through 6.1.8 for comparison.~~ 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<sup>††</sup>, Tables 6.1.1 through 6.1.8 list the bounding maximum  $k_{\text{eff}}$  value, and the associated maximum allowable enrichment. The maximum allowed enrichments and the minimum soluble boron concentrations are ~~defined in Appendix B to the Certificate of Compliance~~ 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.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) loaded with damaged fuel assemblies or a combination of intact and damaged fuel assemblies are listed in Tables 6.1.9 through 6.1.142. The results include the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs qualified for damaged fuel and/or fuel debris (MPC-24E, MPC-24EF, MPC-68, MPC-68F, ~~and MPC-68FF, MPC-32 and MPC-32F~~), Tables 6.1.9 through 6.1.142 indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum  $k_{\text{eff}}$  value, ~~and the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration.~~ For the permissible location of DFCs see Subsection 6.4.4.2. The maximum allowed enrichments are ~~defined in Appendix B to the Certificate of Compliance~~ also listed in Section 2.1.9.

A table listing the maximum  $k_{\text{eff}}$  (including bias, uncertainties, and calculational statistics), calculated  $k_{\text{eff}}$ , standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum  $k_{\text{eff}}$  values for the HI-STORM 100 System are below the limiting design criteria ( $k_{\text{eff}} < 0.95$ ) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

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

Table 6.1.1

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24  
(no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{\text{eff}}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	4.6	0.3080	0.9283	0.9296
14x14B	4.6	---	0.9237	0.9228
14x14C	4.6	---	0.9274	0.9287
14x14D	4.0	---	0.8531	0.8507
14x14E	5.0	---	0.7627	0.7627
15x15A	4.1	---	0.9205	0.9204
15x15B	4.1	---	0.9387	0.9388
15x15C	4.1	---	0.9362	0.9361
15x15D	4.1	---	0.9354	0.9367
15x15E	4.1	---	0.9392	0.9368
15x15F	4.1	0.3648	0.9393 <sup>††</sup>	0.9395 <sup>†††</sup>
15x15G	4.0	---	0.8878	0.8876
15x15H	3.8	---	0.9333	0.9337
16x16A	4.6	0.3447	0.9273	0.9287
17x17A	4.0	0.3243	0.9378	0.9368
17x17B	4.0	---	0.9318	0.9324
17x17C	4.0	---	0.9319	0.9336

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.

<sup>†</sup> The term "maximum  $k_{\text{eff}}$ " as used here, and elsewhere in this document, means the highest possible  $k_{\text{eff}}$ , including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

<sup>††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9383.

<sup>†††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9378.

Table 6.1.2

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24  
WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{\text{eff}}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8884
14x14B	5.0	---	---	0.8900
14x14C	5.0	---	---	0.8950
14x14D	5.0	---	---	0.8518
14x14E	5.0	---	---	0.7132
15x15A	5.0	---	---	0.9119
15x15B	5.0	---	---	0.9284
15x15C	5.0	---	---	0.9236
15x15D	5.0	---	---	0.9261
15x15E	5.0	---	---	0.9265
15x15F	5.0	0.4013	0.9301	0.9314
15x15G	5.0	---	---	0.8939
15x15H	5.0	---	0.9345	0.9366
16x16A	5.0	---	---	0.8955
17x17A	5.0	---	---	0.9264
17x17B	5.0	---	---	0.9284
17x17C	5.0	---	0.9296	0.9294

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.

<sup>†</sup> The term "maximum  $k_{\text{eff}}$ " 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 6.1.3

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF (no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{\text{eff}}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.9380
14x14B	5.0	---	---	0.9312
14x14C	5.0	---	---	0.9356
14x14D	5.0	---	---	0.8875
14x14E	5.0	---	---	0.7651
15x15A	4.5	---	---	0.9336
15x15B	4.5	---	---	0.9465
15x15C	4.5	---	---	0.9462
15x15D	4.5	---	---	0.9440
15x15E	4.5	---	---	0.9455
15x15F	4.5	0.3699	0.9465	0.9468
15x15G	4.5	---	---	0.9054
15x15H	4.2	---	---	0.9423
16x16A	5.0	---	---	0.9341
17x17A	4.4	---	0.9467	0.9447
17x17B	4.4	---	---	0.9421
17x17C	4.4	---	---	0.9433

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.

<sup>†</sup> The term "maximum  $k_{\text{eff}}$ " 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 6.1.4

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF WITH 300 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8963
14x14B	5.0	---	---	0.8974
14x14C	5.0	---	---	0.9031
14x14D	5.0	---	---	0.8588
14x14E	5.0	---	---	0.7249
15x15A	5.0	---	---	0.9161
15x15B	5.0	---	---	0.9321
15x15C	5.0	---	---	0.9271
15x15D	5.0	---	---	0.9290
15x15E	5.0	---	---	0.9309
15x15F	5.0	0.3897	0.9333	0.9332
15x15G	5.0	---	---	0.8972
15x15H	5.0	---	0.9399	0.9399
16x16A	5.0	---	---	0.9021
17x17A	5.0	---	0.9320	0.9332
17x17B	5.0	---	---	0.9316
17x17C	5.0	---	---	0.9312

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.

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

Table 6.1.5

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32  
AND MPC-32F FOR 4.1% ENRICHMENT WITH 1900 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Concentration (ppm)	Maximum <sup>†</sup> $k_{eff}$		
			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
16x16A	4.1	1300	---	---	0.9468
17x17A	4.1	1900	---	---	0.9111
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.

<sup>†</sup> 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 AND MPC-32F FOR 5.0% ENRICHMENT WITH 2600 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )	Minimum Soluble Boron Concentration (ppm)	Maximum $k_{eff}$ <sup>†</sup>		
			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
16x16A	5.0	1900	---	---	0.9474
17x17A	5.0	2600	---	---	0.9161
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.

<sup>†</sup> 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 6.1.7

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 AND MPC-68FF

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7 <sup>††</sup>	---	0.7886	0.7888 <sup>†††</sup>
6x6B <sup>‡</sup>	2.7 <sup>††</sup>	---	0.7833	0.7824 <sup>†††</sup>
6x6C	2.7 <sup>††</sup>	0.2759	0.8024	0.8021 <sup>†††</sup>
7x7A	2.7 <sup>††</sup>	---	0.7963	0.7974 <sup>†††</sup>
7x7B	4.2	0.4061	0.9385	0.9386
8x8A	2.7 <sup>††</sup>	---	0.7690	0.7697 <sup>†††</sup>
8x8B	4.2	0.3934	0.9427	0.9416
8x8C	4.2	0.3714	0.9429	0.9425
8x8D	4.2	---	0.9408	0.9403
8x8E	4.2	---	0.9309	0.9312
8x8F	4.0	---	0.9396	0.9411

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.

<sup>†</sup> 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.

<sup>††</sup> This calculation was performed for 3.0% planar-average enrichment, however, the actual fuel and Certificate of Compliance authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum  $k_{eff}$  value is conservative.

<sup>†††</sup> This calculation was performed for a  $^{10}\text{B}$  loading of 0.0067 g/cm<sup>2</sup>, which is 75% of a minimum  $^{10}\text{B}$  loading of 0.0089 g/cm<sup>2</sup>. The minimum  $^{10}\text{B}$  loading in the MPC-68 is at least 0.037210 g/cm<sup>2</sup>. Therefore, the listed maximum  $k_{eff}$  value is conservative.

<sup>‡</sup> Assemblies in this class contain both MOX and UO<sub>2</sub> pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the Certificate of Compliance specification of authorized contents in Section 2.1.9.

Table 6.1.7 (continued)

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 AND MPC-68FF

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
9x9A	4.2	0.3365	0.9434	0.9417
9x9B	4.2	---	0.9417	0.9436
9x9C	4.2	---	0.9377	0.9395
9x9D	4.2	---	0.9387	0.9394
9x9E	4.0		0.9402	0.9401
9x9F	4.0	---	0.9402	0.9401
9x9G	4.2	---	0.9307	0.9309
10x10A	4.2	0.3379	0.9448 <sup>‡‡</sup>	0.9457*
10x10B	4.2	---	0.9443	0.9436
10x10C	4.2	---	0.9430	0.9433
10x10D	4.0	---	0.9383	0.9376
10x10E	4.0	---	0.9157	0.9185

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.

<sup>†</sup> 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.

<sup>‡‡</sup> KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9451.

\* KENO5a verification calculation resulted in a maximum  $k_{eff}$  of 0.9453.

Table 6.1.8

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum <sup>†</sup> $k_{eff}$		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7 <sup>††</sup>	---	0.7886	0.7888
6x6B <sup>†††</sup>	2.7	---	0.7833	0.7824
6x6C	2.7	0.2759	0.8024	0.8021
7x7A	2.7	---	0.7963	0.7974
8x8A	2.7	---	0.7690	0.7697

Notes:

1. 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.
2. These calculations were performed for a  $^{10}\text{B}$  loading of  $0.0067 \text{ g/cm}^2$ , which is 75% of a minimum  $^{10}\text{B}$  loading of  $0.0089 \text{ g/cm}^2$ . The minimum  $^{10}\text{B}$  loading in the MPC-68F is  $0.010 \text{ g/cm}^2$ . Therefore, the listed maximum  $k_{eff}$  values are conservative.

<sup>†</sup> 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.

<sup>††</sup> These calculations were performed for 3.0% planar-average enrichment, however, the ~~actual fuel and Certificate of Compliance~~ *authorized contents* are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum  $k_{eff}$  values are conservative.

<sup>†††</sup> Assemblies in this class contain both MOX and  $\text{UO}_2$  pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in the ~~Certificate of Compliance~~ *specification of authorized contents in Section 2.1.9.*

Table 6.1.9

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-24E AND MPC-24EF  
WITH UP TO 4 DFCs

Fuel Assembly Class	Maximum Allowable Enrichment (wt% $^{235}\text{U}$ )		Minimum Soluble Boron Concentration (ppm)	Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris		HI-TRAC	HI-STAR
All PWR Classes	4.0	4.0	0	0.9486	0.9480
<i>All PWR Classes</i>	<i>5.0</i>	<i>5.0</i>	<i>600</i>	<i>0.9177</i>	<i>0.9185</i>

Table 6.1.10

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68, MPC-68F AND MPC-68FF  
WITH UP TO 68 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )		Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
6x6A, 6x6B, 6x6C, 7x7A, 8x8A	2.7	2.7	0.8024	0.8021

Table 6.1.11

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-68 AND MPC-68FF  
WITH UP TO 16 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )		Maximum $k_{eff}$	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
All BWR Classes	3.7	4.0	0.9328	0.9328

Table 6.1.12

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

<i>Fuel Assembly Class of Intact Fuel</i>	<i>Maximum Allowable Enrichment for Intact Fuel and Damaged Fuel/Fuel Debris (wt% <sup>235</sup>U)</i>	<i>Minimum Soluble Boron Content (ppm)</i>	<i>Maximum <math>k_{eff}</math></i>	
			<i>HI-TRAC</i>	<i>HI-STAR</i>
<i>14x14A, B, C, D, E</i>	<i>4.1</i>	<i>1500</i>	<i>---</i>	<i>0.9336</i>
	<i>5.0</i>	<i>2300</i>	<i>---</i>	<i>0.9269</i>
<i>15x15A, B, C, G</i>	<i>4.1</i>	<i>1900</i>	<i>0.9349</i>	<i>0.9350</i>
	<i>5.0</i>	<i>2700</i>	<i>---</i>	<i>0.9365</i>
<i>15x15D, E, F, H</i>	<i>4.1</i>	<i>2100</i>	<i>---</i>	<i>0.9340</i>
	<i>5.0</i>	<i>2900</i>	<i>0.9382</i>	<i>0.9397</i>
<i>16x16A</i>	<i>4.1</i>	<i>1500</i>	<i>---</i>	<i>0.9335</i>
	<i>5.0</i>	<i>2300</i>	<i>---</i>	<i>0.9289</i>
<i>17x17A, B, C</i>	<i>4.1</i>	<i>2100</i>	<i>---</i>	<i>0.9294</i>
	<i>5.0</i>	<i>2900</i>	<i>---</i>	<i>0.9367</i>



## 6.2 SPENT FUEL LOADING

Specifications for the BWR and PWR fuel assemblies that were analyzed are given in Tables 6.2.1 and 6.2.2, respectively. For the BWR fuel characteristics, the number and dimensions for the water rods are the actual number and dimensions. For the PWR fuel characteristics, the actual number and dimensions of the control rod guide tubes and thimbles are used. Table 6.2.1 lists 72 unique BWR assemblies while Table 6.2.2 lists 46 unique PWR assemblies, all of which were explicitly analyzed for this evaluation. Examination of Tables 6.2.1 and 6.2.2 reveals that there are a large number of minor variations in fuel assembly dimensions.

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in ~~the Certificate of Compliance~~ *defining the authorized contents* could limit the applicability of the HI-STORM 100 System. To resolve this limitation, bounding criticality analyses are presented in this section for a number of defined fuel assembly classes for both fuel types (PWR and BWR). The results of the bounding criticality analyses justify using bounding fuel dimensions, ~~as defined in the Certificate of Compliance~~ *for defining the authorized contents*.

### 6.2.1 Definition of Assembly Classes

For each array size (e.g., 6x6, 7x7, 15x15, etc.), the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; (3) number and locations of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Tables 6.2.1 and 6.2.2, respectively. It should be noted that these assembly classes are unique to this evaluation and are not known to be consistent with any class designations in the open literature.

For each assembly class, calculations have been performed for all of the dimensional variations for which data is available (i.e., all data in Tables 6.2.1 and 6.2.2). These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

Therefore, for each assembly class, a bounding assembly was defined based on the above characteristics and a calculation for the bounding assembly was performed to demonstrate compliance with the regulatory requirement of  $k_{\text{eff}} < 0.95$ . In some assembly classes this bounding assembly corresponds directly to one of the actual (real) assemblies; while in most

assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, ~~the Certificate of Compliance will define acceptability~~ *the authorized contents in Section 2.1.9 are defined* in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R). Additionally, parametric studies were performed for a PWR assembly (the 15x15F assembly class) in the MPC-24 and MPC-32 with soluble boron in the water flooding the MPC. The results of these studies are shown in Table 6.2.3 through 6.2.5, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod/guide tube thickness, (5) artificially replacing the Zircaloy water rod tubes/guide tubes with water, ~~and~~ (6) maximizing the channel thickness (for BWR Assemblies), *and (7) increasing the active length*. These results, and the many that follow, justify the approach for using bounding dimensions ~~in the Certificate of Compliance for defining the authorized contents~~. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the ~~Certificate of Compliance specification of the authorized contents~~. As these studies were performed with and without soluble boron, they also demonstrate that the bounding dimensions are valid independent of the soluble boron concentration.

As mentioned, the bounding approach used in these analyses often results in a maximum  $k_{\text{eff}}$  value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

## 6.2.2 Intact PWR Fuel Assemblies

### 6.2.2.1 Intact PWR Fuel Assemblies in the MPC-24 without Soluble Boron

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9395). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.20) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.6 through 6.2.22 for the fully flooded condition without soluble boron in the water.

Tables 6.2.6 through 6.2.22 show the maximum  $k_{\text{eff}}$  values for the assembly classes that are acceptable for storage in the MPC-24. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations for the MPC-24 were performed for a  $^{10}\text{B}$  loading of  $0.020 \text{ g/cm}^2$ , which is 75% of the minimum loading, of  $0.0267 \text{ g/cm}^2$  for Boral, or 90% of the minimum loading of  $0.0223 \text{ g/cm}^2$  for Metamic, specified on BM-1478, Bill of Materials for 24 Assembly HI-STAR 100 PWR MPC, in Section 1.5. The maximum allowable enrichment in the MPC-24 varies from 3.8 to 5.0 wt%  $^{235}\text{U}$ , depending on the assembly class, and is defined in Tables 6.2.6 through 6.2.22. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.6 through 6.2.22 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected for the Certificate of Compliance to define the authorized contents and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.6). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.9), the authorized contents dimensions listed in the Certificate of Compliance are based on the assembly dimensions from that single assembly. All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

The results of the analyses for the MPC-24, which were performed for all assemblies in each class (see Tables 6.2.6 through 6.2.22), further confirm the validity of the bounding dimensions established in Section 6.2.1. Thus, for all following calculations, namely analyses of the MPC-24E, MPC-32, and MPC-24 with soluble boron present in the water, only the bounding assembly in each class is analyzed.

#### 6.2.2.2 Intact PWR Fuel Assemblies in the MPC-24 with Soluble Boron

Additionally, the HI-STAR 100 system is designed to allow credit for the soluble boron typically present in the water of PWR spent fuel pools. For a minimum soluble boron concentration of 400ppm, the maximum allowable fuel enrichment is 5.0 wt%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.2 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9366). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

### 6.2.2.3 Intact PWR Assemblies in the MPC-24E and MPC-24EF with and without Soluble Boron

The MPC-24E and MPC-24EF are variations of the MPC-24, which provide for storage of higher enriched fuel than the MPC-24 through optimization of the storage cell layout. The MPC-24E and MPC-24EF also allow for the loading of up to 4 PWR Damaged Fuel Containers (DFC) with damaged PWR fuel (MPC-24E and MPC-24EF) and PWR fuel debris (MPC-24EF only). The requirements for damaged fuel and fuel debris in the MPC-24E and MPC-24EF are 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%  $^{235}\text{U}$ , 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  $k_{\text{eff}}$  for the bounding assembly in the assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15F assembly class at 4.5% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9468). The calculated  $k_{\text{eff}}$  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%  $^{235}\text{U}$  for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.4 shows the maximum  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9399). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

### 6.2.2.4 Intact PWR Assemblies in the MPC-32 and MPC-32F

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

~~For a minimum soluble boron concentration of 1900ppm, the maximum allowable fuel enrichment is of 4.1 wt%  $^{235}\text{U}$  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  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The ~~15x15F~~16x16A assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.94468). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

~~For a minimum soluble boron concentration of 2600ppm, the maximum allowable fuel enrichment is of 5.0 wt%  $^{235}\text{U}$  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  $k_{\text{eff}}$  for the bounding assembly in each assembly class. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit. The 15x15F assembly class has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9483). The calculated  $k_{\text{eff}}$  and calculational uncertainty for each class is listed in Appendix 6.C.

### 6.2.3 Intact BWR Fuel Assemblies in the MPC-68 and MPC-68FF

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 and MPC-68FF are summarized in Tables 6.2.23 through 6.2.40 for the fully flooded condition. In all cases, the gadolinia ( $\text{Gd}_2\text{O}_3$ ) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.23 through 6.2.40 show the maximum  $k_{\text{eff}}$  values for assembly classes that are acceptable for storage in the MPC-68 and MPC-68FF. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 and MPC-68FF were performed with a  $^{10}\text{B}$  loading of  $0.0279 \text{ g/cm}^2$ , which is 75% of the minimum loading, of  $0.0372 \text{ g/cm}^2$  for Boral, or 90% of the minimum loading of  $0.031 \text{ g/cm}^2$  for Metamic, specified on BM-1479, Bill of Materials for 68 Assembly HI-STAR 100 BWR MPC, in Section 1.5. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a  $^{10}\text{B}$  loading of  $0.0067 \text{ g/cm}^2$ . The maximum allowable enrichment in the MPC-68 and MPC-68FF varies from 2.7 to 4.2 wt%  $^{235}\text{U}$ , depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.7 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68 and MPC-68FF.

Tables 6.2.23 through 6.2.40 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected for the Certificate of Compliance to define the authorized contents and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.27), the authorized contents dimensions listed in the Certificate of Compliance are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain

assemblies with thicker channels (e.g., 120 mils), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.23). All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum  $k_{\text{eff}}$  values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the ~~Certificate of Compliance~~ *specification of the authorized contents* has no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the ~~Certificate of Compliance~~ *specification of the authorized contents*. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum  $k_{\text{eff}}$  value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils thick; and (3) the active fuel length of the full length rods is 155 inches. Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

#### 6.2.4 BWR and PWR Damaged Fuel Assemblies and Fuel Debris

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 ~~Section 2.1.3 and Appendix B to the Certificate of Compliance~~ *Table 1.0.1*. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. ~~Four~~ *Five* different DFC types with different cross sections are considered; three types for BWR fuel and ~~one~~ *two* for PWR fuel. DFCs containing fuel debris must be stored in the MPC-68F, MPC-68FF, ~~or MPC-24EF or MPC-32F~~. DFCs containing BWR damaged fuel assemblies may be stored in the MPC-68, MPC-68F or MPC-68FF. DFCs containing PWR damaged fuel may be stored in the MPC-24E, ~~and MPC-24EF, MPC-32 or MPC-32F~~. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4.

#### 6.2.4.1 Damaged BWR Fuel Assemblies and BWR Fuel Debris in Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

Tables 6.2.41 through 6.2.45 show the maximum  $k_{\text{eff}}$  values for the five assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. All maximum  $k_{\text{eff}}$  values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a  $^{10}\text{B}$  loading of  $0.0067 \text{ g/cm}^2$ , which is 75% of a minimum loading,  $0.0089 \text{ g/cm}^2$ . However, because the practical manufacturing lower limit for minimum  $^{10}\text{B}$  loading is  $0.01 \text{ g/cm}^2$ , the minimum  $^{10}\text{B}$  loading of  $0.01 \text{ g/cm}^2$  is specified on ~~BM-1479, Bill of Materials for 68 Assembly HI-STAR 100 BWR MPC, the drawing~~ in Section 1.5, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt%  $^{235}\text{U}$ , while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt%  $^{235}\text{U}$  in the ~~Certificate of Compliance~~ *specification of the authorized contents*. Therefore, the maximum  $k_{\text{eff}}$  values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.41 through 6.2.45 for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt%  $^{235}\text{U}$  enrichment has the highest reactivity (maximum  $k_{\text{eff}}$  of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual  $^{10}\text{B}$  loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a  $^{10}\text{B}$  loading of  $0.0089 \text{ g/cm}^2$ , which conservatively bounds the analysis of damaged BWR fuel assemblies in an MPC-68 or MPC-68FF with a minimum  $^{10}\text{B}$  loading of  $0.0372 \text{ g/cm}^2$ , damaged BWR fuel assemblies may also be stored in the MPC-68 or MPC-68FF. However, fuel debris is limited to the MPC-68F and MPC-68FF by ~~Appendix B to the Certificate of Compliance~~ *the specification of the authorized contents*.

Tables 6.2.41 through 6.2.45 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and  $k_{\text{eff}}$  values in the following rows above the bold double lines, and the bounding dimensions selected ~~for the Certificate of Compliance to define the authorized contents~~ and corresponding bounding  $k_{\text{eff}}$  values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.43), the *authorized contents* dimensions listed ~~in the Certificate of Compliance~~ are based on the assembly dimensions from that single assembly. All of the maximum  $k_{\text{eff}}$  values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

#### 6.2.4.2 Damaged BWR Fuel Assemblies and Fuel Debris in the MPC-68 and MPC-68FF

Damaged BWR fuel assemblies and fuel debris from all BWR classes may be loaded into the MPC-68 and MPC-68FF by restricting the locations of the DFCs to 16 specific cells on the periphery of the fuel basket. The MPC-68 may be loaded with up to 16 DFCs containing damaged fuel assemblies. The MPC-68FF may also be loaded with up to 16 DFCs, with up to 8 DFCs containing fuel debris.

For all assembly classes, the enrichment of the damaged fuel or fuel debris is limited to a maximum of 4.0 wt%  $^{235}\text{U}$ , while the enrichment of the intact assemblies stored together with the damaged fuel is limited to a maximum of 3.7 wt%  $^{235}\text{U}$ . The maximum  $k_{\text{eff}}$  is 0.9328. The criticality evaluation of the damaged fuel assemblies and fuel debris in the MPC-68 and MPC-68FF is presented in Section 6.4.4.2.

#### 6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-24E and MPC-24EF

In addition to storing intact PWR fuel assemblies, the HI-STORM 100 System is designed to store damaged PWR fuel assemblies (MPC-24E, ~~and MPC-24EF~~, *MPC-32 and MPC-32F*) and fuel debris (MPC-24EF *and MPC-32F* only). Damaged fuel assemblies and fuel debris are defined in ~~Section 2.1.3 and Appendix B of the Certificate of Compliance~~ *Table 1.0.1*. Damaged PWR fuel assemblies and fuel debris are required to be loaded into PWR Damaged Fuel Containers (DFCs) ~~prior to being loaded into the MPC~~.

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

Up to four DFCs may be stored in the MPC-24E or MPC-24EF. When loaded with damaged fuel and/or fuel debris, the maximum enrichment for intact and damaged fuel is 4.0 wt%  $^{235}\text{U}$  for all assembly classes listed in Table 6.2.6 through 6.2.22 *without credit for soluble boron*. The maximum  $k_{\text{eff}}$  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%  $^{235}\text{U}$  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 and MPC-32F

*Up to eight DFCs may be stored in the MPC-32 or MPC-32F. For a maximum allowable fuel enrichment of 4.1 wt%  $^{235}\text{U}$  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%  $^{235}\text{U}$  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.*

#### 6.2.5 Thoria Rod Canister

Additionally, the HI-STORM 100 System is designed to store a Thoria Rod Canister in the MPC-68, MPC-68F or MPC-68FF. 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  $^{235}\text{U}$  content of these rods correspond to  $\text{UO}_2$  rods with an initial enrichment of approximately 1.7 wt%  $^{235}\text{U}$ ). 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 6.2.46. The criticality evaluation are presented in Subsection 6.4.6.

Table 6.2.1 (page 1 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694	36	0.5645	0.0350	0.4940	110.0	0	n/a	n/a	0.060	4.290
6x6A02	Zr	0.694	36	0.5645	0.0360	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A03	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A04	Zr	0.694	36	0.5550	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A05	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A06	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6A07	Zr	0.700	36	0.5555	0.03525	0.4780	110.0	0	n/a	n/a	0.060	4.290
6x6A08	Zr	0.710	36	0.5625	0.0260	0.4980	110.0	0	n/a	n/a	0.060	4.290
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B02	Zr	0.694	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B03	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B04	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6B05	Zr	0.710	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6C Assembly Class												
6x6C01	Zr	0.740	36	0.5630	0.0320	0.4880	77.5	0	n/a	n/a	0.060	4.542
7x7A Assembly Class												
7x7A01	Zr	0.631	49	0.4860	0.0328	0.4110	80	0	n/a	n/a	0.060	4.542

Table 6.2.1 (page 2 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738	49	0.5630	0.0320	0.4870	150	0	n/a	n/a	0.080	5.278
7x7B02	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.102	5.291
7x7B03	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.080	5.278
7x7B04	Zr	0.738	49	0.5700	0.0355	0.4880	150	0	n/a	n/a	0.080	5.278
7x7B05	Zr	0.738	49	0.5630	0.0340	0.4775	150	0	n/a	n/a	0.080	5.278
7x7B06	Zr	0.738	49	0.5700	0.0355	0.4910	150	0	n/a	n/a	0.080	5.278
8x8A Assembly Class												
8x8A01	Zr	0.523	64	0.4120	0.0250	0.3580	110	0	n/a	n/a	0.100	4.290
8x8A02	Zr	0.523	63	0.4120	0.0250	0.3580	120	0	n/a	n/a	0.100	4.290

Table 6.2.1 (page 3 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B02	Zr	0.636	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B03	Zr	0.640	63	0.4930	0.0340	0.4160	150	1	0.493	0.425	0.100	5.278
8x8B04	Zr	0.642	64	0.5015	0.0360	0.4195	150	0	n/a	n/a	0.100	5.278
8x8C Assembly Class												
8x8C01	Zr	0.641	62	0.4840	0.0350	0.4050	150	2	0.484	0.414	0.100	5.278
8x8C02	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.000	no channel
8x8C03	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.080	5.278
8x8C04	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C05	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C06	Zr	0.640	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.100	5.278
8x8C07	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C08	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.493	0.425	0.100	5.278
8x8C09	Zr	0.640	62	0.4930	0.0340	0.4160	150	2	0.493	0.425	0.100	5.278
8x8C10	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C11	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.215
8x8C12	Zr	0.636	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.120	5.215

Table 6.2.1 (page 4 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640	60	0.4830	0.0320	0.4110	150	2 large/ 2 small	0.591/ 0.483	0.531/ 0.433	0.100	5.278
8x8D02	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.591	0.531	0.100	5.278
8x8D03	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.483	0.433	0.100	5.278
8x8D04	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.100	5.278
8x8D05	Zr	0.640	60	0.4830	0.0320	0.4100	150	1	1.34	1.26	0.100	5.278
8x8D06	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.120	5.278
8x8D07	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.080	5.278
8x8D08	Zr	0.640	61	0.4830	0.0300	0.4140	150	3	0.591	0.531	0.080	5.278
8x8E Assembly Class												
8x8E01	Zr	0.640	59	0.4930	0.0340	0.4160	150	5	0.493	0.425	0.100	5.278
8x8F Assembly Class												
8x8F01	Zr	0.609	64	0.4576	0.0290	0.3913	150	4 <sup>†</sup>	0.291 <sup>†</sup>	0.228 <sup>†</sup>	0.055	5.390
9x9A Assembly Class												
9x9A01	Zr	0.566	74	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A02	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A03	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.100	5.278
9x9A04	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.120	5.278

<sup>†</sup> Four rectangular water cross segments dividing the assembly into four quadrants

Table 6.2.1 (page 5 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9B Assembly Class												
9x9B01	Zr	0.569	72	0.4330	0.0262	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B02	Zr	0.569	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B03	Zr	0.572	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9C Assembly Class												
9x9C01	Zr	0.572	80	0.4230	0.0295	0.3565	150	1	0.512	0.472	0.100	5.278
9x9D Assembly Class												
9x9D01	Zr	0.572	79	0.4240	0.0300	0.3565	150	2	0.424	0.364	0.100	5.278
9x9E Assembly Class <sup>†</sup>												
9x9E01	Zr	0.572	76	0.4170	0.0265	0.3530	150	5	0.546	0.522	0.120	5.215
9x9E02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215

<sup>†</sup> The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in the CoC Section 2.1.9.

Table 6.2.1 (page 6 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9F Assembly Class*												
9x9F01	Zr	0.572	76	0.4430	0.0285	0.3745	150	5	0.546	0.522	0.120	5.215
9x9F02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215
9x9G Assembly Class												
9x9G01	Zr	0.572	72	0.4240	0.0300	0.3565	150	1	1.668	1.604	0.120	5.278
10x10A Assembly Class												
10x10A01	Zr	0.510	92	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A02	Zr	0.510	78	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A03	Zr	0.510	92/78	0.4040	0.0260	0.3450	155/90	2	0.980	0.920	0.100	5.278
10x10B Assembly Class												
10x10B01	Zr	0.510	91	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B02	Zr	0.510	83	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B03	Zr	0.510	91/83	0.3957	0.0239	0.3413	155/90	1	1.378	1.321	0.100	5.278

\* The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in the CoC Section 2.1.9.

Table 6.2.1 (page 7 of 7)  
 BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
 (all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10C Assembly Class												
10x10C01	Zr	0.488	96	0.3780	0.0243	0.3224	150	5	1.227	1.165	0.055	5.347
10x10D Assembly Class												
10x10D01	SS	0.565	100	0.3960	0.0200	0.3500	83	0	n/a	n/a	0.08	5.663
10x10E Assembly Class												
10x10E01	SS	0.557	96	0.3940	0.0220	0.3430	83	4	0.3940	0.3500	0.08	5.663



Table 6.2.2 (page 1 of 4)  
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.527	0.493	0.0170
14x14A02	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.528	0.490	0.0190
14x14A03	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.526	0.492	0.0170
14x14B Assembly Class											
14x14B01	Zr	0.556	179	0.422	0.0243	0.3659	150	17	0.539	0.505	0.0170
14x14B02	Zr	0.556	179	0.417	0.0295	0.3505	150	17	0.541	0.507	0.0170
14x14B03	Zr	0.556	179	0.424	0.0300	0.3565	150	17	0.541	0.507	0.0170
14x14B04	Zr	0.556	179	0.426	0.0310	0.3565	150	17	0.541	0.507	0.0170
14x14C Assembly Class											
14x14C01	Zr	0.580	176	0.440	0.0280	0.3765	150	5	1.115	1.035	0.0400
14x14C02	Zr	0.580	176	0.440	0.0280	0.3770	150	5	1.115	1.035	0.0400
14x14C03	Zr	0.580	176	0.440	0.0260	0.3805	150	5	1.111	1.035	0.0380
14x14D Assembly Class											
14x14D01	SS	0.556	180	0.422	0.0165	0.3835	144	16	0.543	0.514	0.0145

Table 6.2.2 (page 2 of 4)  
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14E Assembly Class											
14x14E01 <sup>†</sup>	SS	0.453 and 0.411	162 3 8	0.3415 0.3415 0.3415	0.0120 0.0285 0.0200	0.313 0.280 0.297	102	0	n/a	n/a	n/a
14x14E02 <sup>†</sup>	SS	0.453 and 0.411	173	0.3415	0.0120	0.313	102	0	n/a	n/a	n/a
14x14E03 <sup>†</sup>	SS	0.453 and 0.411	173	0.3415	0.0285	0.0280	102	0	n/a	n/a	n/a
15x15A Assembly Class											
15x15A01	Zr	0.550	204	0.418	0.0260	0.3580	150	21	0.533	0.500	0.0165

<sup>†</sup> This is the fuel assembly used at Indian Point 1 (IP-1). This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. It has a different pitch in different sections of the assembly, and different fuel rod dimensions in some rods.

Table 6.2.2 (page 3 of 4)  
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.533	0.499	0.0170
15x15B02	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.546	0.512	0.0170
15x15B03	Zr	0.563	204	0.422	0.0243	0.3660	150	21	0.533	0.499	0.0170
15x15B04	Zr	0.563	204	0.422	0.0243	0.3659	150	21	0.545	0.515	0.0150
15x15B05	Zr	0.563	204	0.422	0.0242	0.3659	150	21	0.545	0.515	0.0150
15x15B06	Zr	0.563	204	0.420	0.0240	0.3671	150	21	0.544	0.514	0.0150
15x15C Assembly Class											
15x15C01	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.493	0.0255
15x15C02	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.511	0.0165
15x15C03	Zr	0.563	204	0.424	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15C04	Zr	0.563	204	0.417	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15D Assembly Class											
15x15D01	Zr	0.568	208	0.430	0.0265	0.3690	150	17	0.530	0.498	0.0160
15x15D02	Zr	0.568	208	0.430	0.0265	0.3686	150	17	0.530	0.498	0.0160
15x15D03	Zr	0.568	208	0.430	0.0265	0.3700	150	17	0.530	0.499	0.0155
15x15D04	Zr	0.568	208	0.430	0.0250	0.3735	150	17	0.530	0.500	0.0150
15x15E Assembly Class											
15x15E01	Zr	0.568	208	0.428	0.0245	0.3707	150	17	0.528	0.500	0.0140
15x15F Assembly Class											
15x15F01	Zr	0.568	208	0.428	0.0230	0.3742	150	17	0.528	0.500	0.0140

Table 6.2.2 (page 4 of 4)  
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS  
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563	204	0.422	0.0165	0.3825	144	21	0.543	0.514	0.0145
15x15H Assembly Class											
15x15H01	Zr	0.568	208	0.414	0.0220	0.3622	150	17	0.528	0.500	0.0140
16x16A Assembly Class											
16x16A01	Zr	0.506	236	0.382	0.0250	0.3255	150	5	0.980	0.900	0.0400
16x16A02	Zr	0.506	236	0.382	0.0250	0.3250	150	5	0.980	0.900	0.0400
17x17A Assembly Class											
<del>17x17A01</del>	<del>Zr</del>	<del>0.496</del>	<del>264</del>	<del>0.360</del>	<del>0.0225</del>	<del>0.3088</del>	<del>144</del>	<del>25</del>	<del>0.474</del>	<del>0.442</del>	<del>0.0160</del>
17x17A02	Zr	0.496	264	0.360	0.0225	0.3088	150	25	0.474	0.442	0.0160
17x17A03	Zr	0.496	264	0.360	0.0250	0.3030	150	25	0.480	0.448	0.0160
17x17B Assembly Class											
17x17B01	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.482	0.450	0.0160
17x17B02	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.474	0.442	0.0160
17x17B03	Zr	0.496	264	0.376	0.0240	0.3215	150	25	0.480	0.448	0.0160
17x17B04	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.427	0.399	0.0140
17x17B05	Zr	0.496	264	0.374	0.0240	0.3195	150	25	0.482	0.450	0.0160
17x17B06	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.480	0.452	0.0140
17x17C Assembly Class											
17x17C01	Zr	0.502	264	0.379	0.0240	0.3232	150	25	0.472	0.432	0.0200
17x17C02	Zr	0.502	264	0.377	0.0220	0.3252	150	25	0.472	0.432	0.0200

Table 6.2.3  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-68  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	water rod thickness	channel thickness
8x8C04 (GE8x8R)	reference	0.9307	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
increase pellet OD (+0.001)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.411	0.030	0.100
decrease pellet OD (-0.001)	-0.0008	0.9299	0.0009	0.483	0.419	0.032	0.409	0.030	0.100
increase clad ID (+0.004)	+0.0027	0.9334	0.0007	0.483	0.423	0.030	0.410	0.030	0.100
decrease clad ID (-0.004)	-0.0034	0.9273	0.0007	0.483	0.415	0.034	0.410	0.030	0.100
increase clad OD (+0.004)	-0.0041	0.9266	0.0008	0.487	0.419	0.034	0.410	0.030	0.100
decrease clad OD (-0.004)	+0.0023	0.9330	0.0007	0.479	0.419	0.030	0.410	0.030	0.100
increase water rod thickness (+0.015)	-0.0019	0.9288	0.0008	0.483	0.419	0.032	0.410	0.045	0.100
decrease water rod thickness (-0.015)	+0.0001	0.9308	0.0008	0.483	0.419	0.032	0.410	0.015	0.100
remove water rods (i.e., replace the water rod tubes with water)	+0.0021	0.9328	0.0008	0.483	0.419	0.032	0.410	0.000	0.100
remove channel	-0.0039	0.9268	0.0009	0.483	0.419	0.032	0.410	0.030	0.000
increase channel thickness (+0.020)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.410	0.030	0.120
<i>reduced active length (120 Inches)</i>	<i>-0.0007</i>	<i>0.9300</i>	<i>0.0007</i>	<i>0.483</i>	<i>0.419</i>	<i>0.032</i>	<i>0.410</i>	<i>0.030</i>	<i>0.100</i>
<i>reduced active length (90 Inches)</i>	<i>-0.0043</i>	<i>0.9264</i>	<i>0.0007</i>	<i>0.483</i>	<i>0.419</i>	<i>0.032</i>	<i>0.410</i>	<i>0.030</i>	<i>0.100</i>

Table 6.2.4  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC 24 with 400ppm soluble boron concentration  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9271	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	-0.0008	0.9263	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	-0.0002	0.9269	0.0005	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0040	0.9311	0.0005	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0033	0.9238	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0042	0.9229	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0035	0.9306	0.0005	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0008	0.9263	0.0005	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	+0.0006	0.9277	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	+0.0028	0.9299	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	-0.0318	0.8953	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2.5  
 REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-32 with 2600ppm soluble boron concentration  
 (all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9389	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	+0.0019	0.9408	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	0.0000	0.9389	0.0004	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0015	0.9404	0.0004	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0015	0.9374	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0002	0.9387	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0007	0.9397	0.0004	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0003	0.9387	0.0004	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	-0.0005	0.9384	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	-0.0005	0.9385	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	+0.0039	0.9428	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2. 6  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14A (4.6% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14A01	0.9295	0.9252	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
14x14A02	0.9286	0.9242	0.0008	0.400	0.3514	0.0243	0.3444	150	0.019
14x14A03	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
Dimensions Listed for Authorized Contents				0.400 (min.)	0.3514 (max.)		0.3444 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (14x14A03)	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017



Table 6.2.7  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14B (4.6% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14B01	0.9159	0.9117	0.0007	0.422	0.3734	0.0243	0.3659	150	0.017
14x14B02	0.9169	0.9126	0.0008	0.417	0.3580	0.0295	0.3505	150	0.017
14x14B03	0.9110	0.9065	0.0009	0.424	0.3640	0.0300	0.3565	150	0.017
14x14B04	0.9084	0.9039	0.0009	0.426	0.3640	0.0310	0.3565	150	0.017
Dimensions Listed for <i>Authorized Contents</i>				0.417 (min.)	0.3734 (max.)		0.3659 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (B14x14B01)	0.9228	0.9185	0.0008	0.417	0.3734	0.0218	0.3659	150	0.017

Table 6.2.8  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14C (4.6% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 176 fuel rods, 5 guide tubes, pitch=0.580, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9258	0.9215	0.0008	0.440	0.3840	0.0280	0.3765	150	0.040
14x14C02	0.9265	0.9222	0.0008	0.440	0.3840	0.0280	0.3770	150	0.040
14x14C03	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038
Dimensions Listed for <i>Authorized Contents</i>				0.440 (min.)	0.3880 (max.)		0.3805 (max.)	150 (max.)	0.038 (min.)
bounding dimensions (14x14C03)	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038

Table 6.2.9  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14D ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14D (4.0% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
180 fuel rods, 16 guide tubes, pitch=0.556, SS clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14D01	0.8507	0.8464	0.0008	0.422	0.3890	0.0165	0.3835	144	0.0145
<i>Dimensions Listed for Authorized Contents</i>				0.422 (min.)	0.3890 (max.)		0.3835 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.10  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 14X14E ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

14x14E (5.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
173 fuel rods, 0 guide tubes, pitch=0.453 and 0.441, SS clad <sup>†</sup>									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length <sup>††</sup>	guide tube thickness
14x14E01	0.7598	0.7555	0.0008	0.3415	0.3175 0.2845 0.3015	0.0120 0.0285 0.0200	0.3130 0.2800 0.2970	102	0.0000
14x14E02	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000
14x14E03	0.6952	0.6909	0.0008	0.3415	0.2845	0.0285	0.2800	102	0.0000
<i>Dimensions Listed for Authorized Contents</i>				0.3415 (min.)	0.3175 (max.)		0.3130 (max.)	102 (max.)	0.0000 (min.)
Bounding dimensions (14x14E02)	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000

<sup>†</sup> This is the IP-1 fuel assembly at Indian Point. This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. Fuel rod dimensions are bounding for each of the three types of rods found in the IP-1 fuel assembly.

<sup>††</sup> Calculations were conservatively performed for a fuel length of 150 inches.

Table 6.2.11  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15A (4.1% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
204 fuel rods, 21 guide tubes, pitch=0.550, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15A01	0.9204	0.9159	0.0009	0.418	0.3660	0.0260	0.3580	150	0.0165
<i>Dimensions Listed for Authorized Contents</i>				0.418 (min.)	0.3660 (max.)		0.3580 (max.)	150 (max.)	0.0165 (min.)

Table 6.2.12  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15B (4.1% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15B01	0.9369	0.9326	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B02	0.9338	0.9295	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B03	0.9362	0.9318	0.0008	0.422	0.3734	0.0243	0.3660	150	0.017
15x15B04	0.9370	0.9327	0.0008	0.422	0.3734	0.0243	0.3659	150	0.015
15x15B05	0.9356	0.9313	0.0008	0.422	0.3736	0.0242	0.3659	150	0.015
15x15B06	0.9366	0.9324	0.0007	0.420	0.3720	0.0240	0.3671	150	0.015
Dimensions Listed for Authorized Contents				0.420 (min.)	0.3736 (max.)		0.3671 (max.)	150 (max.)	0.015 (min.)
bounding dimensions (B15x15B01)	0.9388	0.9343	0.0009	0.420	0.3736	0.0232	0.3671	150	0.015

Table 6.2.13  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15C (4.1% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15C01	0.9255	0.9213	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0255
15x15C02	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0165
15x15C03	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3565	150	0.0165
15x15C04	0.9311	0.9268	0.0008	0.417	0.3570	0.0300	0.3565	150	0.0165
Dimensions Listed for <i>Authorized Contents</i>				0.417 (min.)	0.3640 (max.)		0.3570 (max.)	150 (max.)	0.0165 (min.)
bounding dimensions (B15x15C01)	0.9361	0.9316	0.0009	0.417	0.3640	0.0265	0.3570	150	0.0165

Table 6.2.14  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15D ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15D (4.1% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15D01	0.9341	0.9298	0.0008	0.430	0.3770	0.0265	0.3690	150	0.0160
15x15D02	0.9367	0.9324	0.0008	0.430	0.3770	0.0265	0.3686	150	0.0160
15x15D03	0.9354	0.9311	0.0008	0.430	0.3770	0.0265	0.3700	150	0.0155
15x15D04	0.9339	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150
Dimensions Listed for Authorized Contents				0.430 (min.)	0.3800 (max.)		0.3735 (max.)	150 (max.)	0.0150 (min.)
bounding dimensions (15x15D04)	0.9339 <sup>†</sup>	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150

<sup>†</sup> The  $k_{\text{eff}}$  value listed for the 15x15D02 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9367 (15x15D02) value is listed in Table 6.1.1 as the maximum.



Table 6.2.15  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15E ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15E (4.1% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15E01	0.9368	0.9325	0.0008	0.428	0.3790	0.0245	0.3707	150	0.0140
<i>Dimensions Listed for Authorized Contents</i>				0.428 (min.)	0.3790 (max.)		0.3707 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.16  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15F ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15F (4.1% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15F01	0.9395 <sup>†</sup>	0.9350	0.0009	0.428	0.3820	0.0230	0.3742	150	0.0140
<i>Dimensions Listed for Authorized Contents</i>				0.428 (min.)	0.3820 (max.)		0.3742 (max.)	150 (max.)	0.0140 (min.)

<sup>†</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9383.

Table 6.2.17  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 15X15G ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15G (4.0% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
204 fuel rods, 21 guide tubes, pitch=0.563, SS clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15G01	0.8876	0.8833	0.0008	0.422	0.3890	0.0165	0.3825	144	0.0145
<i>Dimensions Listed for Authorized Contents</i>				0.422 (min.)	0.3890 (max.)		0.3825 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.18  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

15x15H (3.8% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9337	0.9292	0.0009	0.414	0.3700	0.0220	0.3622	150	0.0140
<i>Dimensions Listed for Authorized Contents</i>				0.414 (min.)	0.3700 (max.)		0.3622 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.19  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

16x16A (4.6% Enrichment, Boron-fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> ) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400
16x16A02	0.9263	0.9221	0.0007	0.382	0.3320	0.0250	0.3250	150	0.0400
Dimensions Listed for Authorized Contents				0.382 (min.)	0.3320 (max.)		0.3255 (max.)	150 (max.)	0.0400 (min.)
bounding dimensions (16x16A01)	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400

Table 6.2.20  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17A (4.0% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17A01	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	144	0.016
17x17A021	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016
17x17A032	0.9329	0.9286	0.0008	0.360	0.3100	0.0250	0.3030	150	0.016
Dimensions Listed for Authorized Contents				0.360 (min.)	0.3150 (max.)		0.3088 (max.)	150 (max.)	0.016 (min.)
bounding dimensions (17x17A021)	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016

Table 6.2.21  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17B (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9288	0.9243	0.0009	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B02	0.9290	0.9247	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B03	0.9243	0.9199	0.0008	0.376	0.3280	0.0240	0.3215	150	0.016
17x17B04	0.9324	0.9279	0.0009	0.372	0.3310	0.0205	0.3232	150	0.014
17x17B05	0.9266	0.9222	0.0008	0.374	0.3260	0.0240	0.3195	150	0.016
17x17B06	0.9311	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014
Dimensions Listed for Authorized Contents				0.372 (min.)	0.3310 (max.)		0.3232 (max.)	150 (max.)	0.014 (min.)
bounding dimensions (17x17B06)	0.9311 <sup>†</sup>	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014

<sup>†</sup> The  $k_{\text{eff}}$  value listed for the 17x17B04 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9324 (17x17B04) value is listed in Table 6.1.1 as the maximum.

Table 6.2.22  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24  
 (all dimensions are in inches)

17x17C (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.02 g/cm <sup>2</sup> )									
264 fuel rods, 25 guide tubes, pitch=0.502, Zr clad									
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9293	0.9250	0.0008	0.379	0.3310	0.0240	0.3232	150	0.020
17x17C02	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020
Dimensions Listed for Authorized Contents				0.377 (min.)	0.3330 (max.)		0.3252 (max.)	150 (max.)	0.020 (min.)
bounding dimensions (17x17C02)	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020



Table 6.2.23  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

7x7B (4.2% Enrichment, Boral fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
49 fuel rods, 0 water rods, pitch=0.738, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630	0.4990	0.0320	0.4870	150	n/a	0.080
7x7B02	0.9301	0.9260	0.0007	0.5630	0.4890	0.0370	0.4770	150	n/a	0.102
7x7B03	0.9313	0.9271	0.0008	0.5630	0.4890	0.0370	0.4770	150	n/a	0.080
7x7B04	0.9311	0.9270	0.0007	0.5700	0.4990	0.0355	0.4880	150	n/a	0.080
7x7B05	0.9350	0.9306	0.0008	0.5630	0.4950	0.0340	0.4775	150	n/a	0.080
7x7B06	0.9298	0.9260	0.0006	0.5700	0.4990	0.0355	0.4910	150	n/a	0.080
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4910 (max.)	150 (max.)	n/a	0.120 (max.)
bounding dimensions (B7x7B01)	0.9375	0.9332	0.0008	0.5630	0.4990	0.0320	0.4910	150	n/a	0.102
bounding dimensions with 120 mil channel (B7x7B02)	0.9386	0.9344	0.0007	0.5630	0.4990	0.0320	0.4910	150	n/a	0.120

Table 6.2.24  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

8x8B (4.2% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )												
63 or 64 fuel rods <sup>†</sup> , 1 or 0 water rods <sup>†</sup> , pitch <sup>†</sup> = 0.636-0.642, Zr clad												
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	Fuel rods	Pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	63	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B02	0.9227	0.9185	0.0007	63	0.636	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B03	0.9299	0.9257	0.0008	63	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8B04	0.9236	0.9194	0.0008	64	0.642	0.5015	0.4295	0.0360	0.4195	150	n/a	0.100
Dimensions Listed for Authorized Contents				63 or 64	0.636-0.642	0.4840 (min.)	0.4295 (max.)		0.4195 (max.)	150 (max.)	0.034	0.120 (max.)
bounding (pitch=0.636) (B8x8B01)	0.9346	0.9301	0.0009	63	0.636	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.640) (B8x8B02)	0.9385	0.9343	0.0008	63	0.640	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.642) (B8x8B03)	0.9416	0.9375	0.0007	63	0.642	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120

<sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.25  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

8x8C (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )											
62 fuel rods, 2 water rods, pitch <sup>†</sup> = 0.636-0.641, Zr clad											
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8C02	0.9313	0.9268	0.0009	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.000
8x8C03	0.9329	0.9286	0.0008	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.800
8x8C04	0.9348 <sup>††</sup>	0.9307	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.100
8x8C05	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.120
8x8C06	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8C07	0.9314	0.9273	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.100
8x8C08	0.9339	0.9298	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.034	0.100
8x8C09	0.9301	0.9260	0.0007	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8C10	0.9317	0.9275	0.0008	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C11	0.9328	0.9287	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C12	0.9285	0.9242	0.0008	0.636	0.4830	0.4190	0.0320	0.4110	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.636-0.641	0.4830 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
Bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120

<sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch.

<sup>††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9343.

Table 6.2.26  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

8x8D (4.2% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
60-61 fuel rods, 1-4 water rods <sup>†</sup> , pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830	0.4190	0.0320	0.4110	150	0.03/0.025	0.100
8x8D02	0.9325	0.9284	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8D03	0.9351	0.9309	0.0008	0.4830	0.4190	0.0320	0.4110	150	0.025	0.100
8x8D04	0.9338	0.9296	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.100
8x8D05	0.9339	0.9294	0.0009	0.4830	0.4190	0.0320	0.4100	150	0.040	0.100
8x8D06	0.9365	0.9324	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.120
8x8D07	0.9341	0.9297	0.0009	0.4830	0.4190	0.0320	0.4110	150	0.040	0.080
8x8D08	0.9376	0.9332	0.0009	0.4830	0.4230	0.0300	0.4140	150	0.030	0.080
Dimensions Listed for Authorized Contents				0.4830 (min.)	0.4230 (max.)		0.4140 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B8x8D01)	0.9403	0.9363	0.0007	0.4830	0.4230	0.0300	0.4140	150	0.000	0.120

<sup>†</sup> Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.

Table 6.2.27  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

8x8E (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
59 fuel rods, 5 water rods, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
<i>Dimensions Listed for Authorized Contents</i>				0.4930 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.034 (min.)	0.100 (max.)

Table 6.2.28  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

8x8F (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> ) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9411	0.9366	0.0009	0.4576	0.3996	0.0290	0.3913	150	0.0315	0.055
<i>Dimensions Listed for Authorized Contents</i>				0.4576 (min.)	0.3996 (max.)		0.3913 (max.)	150 (max.)	0.0315 (min.)	0.055 (max.)

Table 6.2.29  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

9x9A (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
74/66 fuel rods <sup>†</sup> , 2 water rods, pitch=0.566, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400	0.3840	0.0280	0.3760	150/90	0.030	0.100
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400	0.3840	0.0280	0.3760	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.4400 (min.)	0.3840 (max.)		0.3760 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.000	0.120

<sup>†</sup> This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

Table 6.2.30  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

9x9B (4.2% Enrichment, Boral <sup>†</sup> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0279 g/cm <sup>2</sup> )											
72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 to 0.572 <sup>†</sup> , Zr clad											
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9380	0.9336	0.0008	0.569	0.4330	0.3807	0.0262	0.3737	150	0.0285	0.100
9x9B02	0.9373	0.9329	0.0009	0.569	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
9x9B03	0.9417	0.9374	0.0008	0.572	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
Dimensions Listed for Authorized Contents				0.572	0.4330 (min.)	0.3810 (max.)		0.3740 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B9x9B01)	0.9436	0.9394	0.0008	0.572	0.4330	0.3810	0.0260	0.3740 <sup>††</sup>	150	0.000	0.120

<sup>†</sup> This assembly class was analyzed and qualified for a small variation in the pitch.

<sup>††</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.



Table 6.2.31  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

9x9C (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
80 fuel rods, 1 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230	0.3640	0.0295	0.3565	150	0.020	0.100
<i>Dimensions Listed for Authorized Contents</i>				0.4230 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.020 (min.)	0.100 (max.)

Table 6.2.32  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

9x9D (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
79 fuel rods, 2 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.9394	0.9350	0.0009	0.4240	0.3640	0.0300	0.3565	150	0.0300	0.100
<i>Dimensions Listed for Authorized Contents</i>				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0300 (min.)	0.100 (max.)

Table 6.2.33  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

9x9E (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.9334	0.9293	0.0007	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
9x9E02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents <sup>†</sup>				0.4170 (min.)	0.3640 (max.)		0.3530 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9E02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

<sup>†</sup> This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed in the Certificate of Compliance for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Certificate of Compliance Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.34  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

9x9F (4.0% Enrichment, Borafixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.9307	0.9265	0.0007	0.4430	0.3860	0.0285	0.3745	150	0.0120	0.120
9x9F02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents <sup>†</sup>				0.4430 (min.)	0.3860 (max.)		0.3745 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9F02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

<sup>†</sup> This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed in the Certificate of Compliance for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Certificate of Compliance Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.35  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 9X9G ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

9x9G (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9G01	0.9309	0.9265	0.0008	0.4240	0.3640	0.0300	0.3565	150	0.0320	0.120
<i>Dimensions Listed for Authorized Contents</i>				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0320 (min.)	0.120 (max.)

Table 6.2.36  
 MAXIMUM  $k_{\text{EFF}}$  VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10A (4.2% Enrichment, Boral <sup>†</sup> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0279 g/cm <sup>2</sup> )										
92/78 fuel rods <sup>†</sup> , 2 water rods, pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040	0.3520	0.0260	0.3450	155/90	0.030	0.100
Dimensions Listed for <i>Authorized Contents</i>				0.4040 (min.)	0.3520 (max.)		0.3455 (max.)	150 <sup>††</sup> (max.)	0.030 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457 <sup>†††</sup>	0.9414	0.0008	0.4040	0.3520	0.0260	0.3455 <sup>‡</sup>	155	0.030	0.120

<sup>†</sup> This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

<sup>††</sup> Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the ~~Certificate of Compliance~~ *specification for the authorized contents* limits the active fuel length to 150 inches. This is due to the fact that the ~~Boral~~ *fixed neutron absorber* panels are 156 inches in length.

<sup>†††</sup> KENO5a verification calculation resulted in a maximum  $k_{\text{eff}}$  of 0.9453.

<sup>‡</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.37  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF  
 (all dimensions are in inches)

10x10B (4.2% Enrichment, <del>Borated</del> <i>fixed neutron absorber</i> $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
91/83 fuel rods <sup>†</sup> , 1 water rods (square, replacing 9 fuel rods), pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957	0.3480	0.0239	0.3413	155/90	0.0285	0.100
Dimensions Listed for <i>Authorized Contents</i>				0.3957 (min.)	0.3480 (max.)		0.3420 (max.)	150 <sup>††</sup> (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957	0.3480	0.0239	0.3420 <sup>†††</sup>	155	0.000	0.120

<sup>†</sup> This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

<sup>††</sup> Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the ~~Certificate of Compliance~~ *specification for the authorized contents* limits the active fuel length to 150 inches. This is due to the fact that the ~~Borated~~ *fixed neutron absorber* panels are 156 inches in length.

<sup>†††</sup> This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.38  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10C (4.2% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.9433	0.9392	0.0007	0.3780	0.3294	0.0243	0.3224	150	0.031	0.055
Dimensions Listed for Authorized Contents				0.3780 (min.)	0.3294 (max.)		0.3224 (max.)	150 (max.)	0.031 (min.)	0.055 (max.)



Table 6.2.39  
 MAXIMUM  $K_{\text{EFF}}$  VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10D (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
100 fuel rods, 0 water rods, pitch=0.565, SS clad										
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960	0.3560	0.0200	0.350	83	n/a	0.080
Dimensions Listed for Authorized Contents				0.3960 (min.)	0.3560 (max.)		0.350 (max.)	83 (max.)	n/a	0.080 (max.)

Table 6.2.40  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10E (4.0% Enrichment, Boron fixed neutron absorber $^{10}\text{B}$ minimum loading of 0.0279 g/cm <sup>2</sup> )										
96 fuel rods, 4 water rods, pitch=0.557, SS clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940	0.3500	0.0220	0.3430	83	0.022	0.080
<i>Dimensions Listed for Authorized Contents</i>				0.3940 (min.)	0.3500 (max.)		0.3430 (max.)	83 (max.)	0.022 (min.)	0.080 (max.)

Table 6.2.41  
 MAXIMUM K<sub>EFF</sub> VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF  
 (all dimensions are in inches)

6x6A (3.0% Enrichment <sup>†</sup> , <del>Boral</del> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )												
35 or 36 fuel rods <sup>††</sup> , 1 or 0 water rods <sup>††</sup> , pitch <sup>††</sup> =0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum k <sub>eff</sub>	calculated k <sub>eff</sub>	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4940	110	n/a	0.060
6x6A02	0.7517	0.7476	0.0007	0.694	36	0.5645	0.4925	0.0360	0.4820	110	n/a	0.060
6x6A03	0.7545	0.7501	0.0008	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6A04	0.7537	0.7494	0.0008	0.694	36	0.5550	0.4850	0.0350	0.4820	110	n/a	0.060
6x6A05	0.7555	0.7512	0.0008	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6A06	0.7618	0.7576	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6A07	0.7588	0.7550	0.0005	0.700	36	0.5555	0.4850	0.03525	0.4780	110	n/a	0.060
6x6A08	0.7808	0.7766	0.0007	0.710	36	0.5625	0.5105	0.0260	0.4980	110	n/a	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5550 (min.)	0.5105 (max.)	0.02225	0.4980 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6A01)	0.7727	0.7685	0.0007	0.694	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A02)	0.7782	0.7738	0.0008	0.700	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A03)	0.7888	0.7846	0.0007	0.710	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

<sup>††</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.42  
 MAXIMUM  $k_{\text{EFF}}$  VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF  
 (all dimensions are in inches)

6x6B (3.0% Enrichment <sup>†</sup> , <del>Boral</del> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )												
35 or 36 fuel rods <sup>††</sup> (up to 9 MOX rods), 1 or 0 water rods <sup>††</sup> , pitch <sup>††</sup> =0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum $k_{\text{eff}}$	calculated $k_{\text{eff}}$	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	0.7604	0.7563	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6B02	0.7618	0.7577	0.0007	0.694	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B03	0.7619	0.7578	0.0007	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B04	0.7686	0.7644	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6B05	0.7824	0.7785	0.0006	0.710	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5625 (min.)	0.4945 (max.)		0.4820 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6B01)	0.7822 <sup>†††</sup>	0.7783	0.0006	0.710	35	0.5625	0.4945	0.0340	0.4820	120	0.0	0.060

Note:

1. These assemblies contain up to 9 MOX pins. The composition of the MOX fuel pins is given in Table 6.3.4.

<sup>†</sup> The <sup>235</sup>U enrichment of the MOX and UO<sub>2</sub> pins is assumed to be 0.711% and 3.0%, respectively.

<sup>††</sup> This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

<sup>†††</sup> The  $k_{\text{eff}}$  value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1 $\sigma$ ). Therefore, the 0.7824 value is listed in Tables 6.1.7 and 6.1.8 as the maximum.

Table 6.2.43  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

6x6C (3.0% Enrichment <sup>†</sup> , <del>Borated</del> <i>fixed neutron absorber</i> <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )										
36 fuel rods, 0 water rods, pitch=0.740, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630	0.4990	0.0320	0.4880	77.5	n/a	0.060
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4880 (max.)	77.5 (max.)	n/a	0.060 (max.)

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents to 2.7%*.

Table 6.2.44  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

7x7A (3.0% Enrichment <sup>†</sup> , <del>Boral</del> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )										
49 fuel rods, 0 water rods, pitch=0.631, Zr clad										
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.7974	0.7932	0.0008	0.4860	0.4204	0.0328	0.4110	80	n/a	0.060
Dimensions Listed for Authorized Contents				0.4860 (min.)	0.4204 (max.)		0.4110 (max.)	80 (max.)	n/a	0.060 (max.)

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the Certificate of Compliance specification for the authorized contents to 2.7%.

Table 6.2.45  
 MAXIMUM  $K_{EFF}$  VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

8x8A (3.0% Enrichment <sup>†</sup> , <del>Boral</del> fixed neutron absorber <sup>10</sup> B minimum loading of 0.0067 g/cm <sup>2</sup> )											
63 or 64 fuel rods <sup>††</sup> , 0 water rods, pitch=0.523, Zr clad											
Fuel Assembly Designation	maximum $k_{eff}$	calculated $k_{eff}$	standard deviation	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	64	0.4120	0.3620	0.0250	0.3580	110	n/a	0.100
8x8A02	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100
Dimensions Listed for Authorized Contents				63	0.4120 (min.)	0.3620 (max.)		0.3580 (max.)	120 (max.)	n/a	0.100 (max.)
bounding dimensions (8x8A02)	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100

<sup>†</sup> Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

<sup>††</sup> This assembly class was analyzed and qualified for a variation in the number of fuel rods.

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 <sub>2</sub> and 98.2% ThO <sub>2</sub>
Initial Enrichment	93.5 wt% <sup>235</sup> U for 1.8% of the fuel
Maximum k <sub>eff</sub>	0.1813
Calculated k <sub>eff</sub>	0.1779
Standard Deviation	0.0004



## 6.3 MODEL SPECIFICATION

### 6.3.1 Description of Calculational Model

Figures 6.3.1, 6.3.1.a, 6.3.2 and 6.3.3 show representative horizontal cross sections of the four types of cells used in the calculations, and Figures 6.3.4 through 6.3.6 illustrate the basket configurations used. Four different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket
- an optimized 24 PWR assembly basket (24E / 24EF)
- a 32 PWR assembly basket
- a 68 BWR assembly basket.

For all four basket designs, the same techniques and the same level of detail are used in the calculational models.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7. Although the ~~Boral~~fixed neutron absorber panels are 156 inches in length, which is much longer than the active fuel length (maximum of 150 inches), they are assumed equal to *or less than* the active fuel length in the calculations. As shown on the Drawings in Section 1.5, 16 of the 24 periphery ~~Boral~~fixed neutron absorber panels on the MPC-24 and MPC-24E/EF have reduced width (i.e., 6.25 inches wide as opposed to 7.5 inches). However, the calculational models for these baskets conservatively assume all of the periphery ~~Boral~~fixed neutron absorber panels are 6.25 inches in width. *Note that Figures 6.3.1 through 6.3.3 show Boral as the fixed neutron absorber. The effect of using Metamic as fixed neutron absorber is discussed in Subsection 6.4.11.*

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design conditions important to criticality safety (see Subsection 6.4.2.5), and thus from a criticality standpoint, the normal, off-normal, and accident conditions are identical and do not require individual models.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and ~~Boral~~fixed neutron absorber panels on the stainless steel walls of the storage cells. Under the conditions of storage, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ( $k_{\text{eff}} < 0.52$ ). For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel

rod pellet-to-clad gaps. Appendix 6.D provides sample input files for two of the MPC basket designs (MPC-68 and MPC-24) in the HI-STORM 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included. Furthermore, the water above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

As indicated in Figures 6.3.1 through 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO-3 and MCNP4a were used to determine the direction of the manufacturing tolerances, which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity.

CASMO-3 was used for one of each of the two principal basket designs, i.e. for the flux trap design MPC-24 and for the non-fluxtrap design MPC-68. The effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. Generally, the conclusions in Table 6.3.1 are directly applicable to the MPC-24E/EF and the MPC-32. Exceptions are the conclusions for the water temperature and void percentage, which are not directly applicable to the MPC-32 due to the presence of high soluble boron concentrations in this canister. This condition is addressed in Section 6.4.2.1 where the optimum moderation is determined for the MPC-32.

Additionally, MCNP4a calculations are performed to evaluate the tolerances of the various basket dimensions of the MPC-68, MPC-24 and MPC-32 in further detail. The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a. Calculated  $k_{\text{eff}}$  results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR assemblies. Each of the basket dimensions are evaluated for their minimum, nominal and maximum values from the Drawings of section 1.5. For PWR MPC designs, the reactivity effect of tolerances with soluble boron present in the water is additionally determined. Due to the close similarity between the MPC-24 and MPC-24E, the basket dimensions are only evaluated for the MPC-24, and the same dimensional assumptions are applied to both MPC designs.

Based on the MCNP4a and CASMO-3 calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined. Because the reactivity effect (positive or negative) of the manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

As demonstrated in this section, design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, the fixed neutron absorbing panels (~~Boral~~) and the soluble boron concentration in the water during loading/unloading operations. As shown in Chapter 11, none of these parameters are affected during any of the design basis off-normal or accident conditions involving handling, packaging, transfer or storage.

### 6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM 100 System are listed in Table 6.3.4.

The HI-STORM 100 System is designed such that the fixed neutron absorber (~~Boral~~) will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of ~~Boral~~ *fixed neutron absorber* are provided in Section 1.2.1.3.1.

The continued efficacy of the ~~Boral~~ *fixed neutron absorber* is assured by acceptance testing, documented in Section 9.1.5.3, to validate the  $^{10}\text{B}$  (poison) concentration in the ~~Boral~~ *fixed neutron absorber*. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an MCNP4a calculation of the number of neutrons absorbed in the  $^{10}\text{B}$  was performed. The calculation conservatively assumed a constant neutron source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of  $^{10}\text{B}$  atoms destroyed is only 2.6E-09 in 50 years. Thus, the reduction in  $^{10}\text{B}$  concentration in the ~~Boral~~ *fixed neutron absorber* by neutron absorption is negligible. In addition, ~~analysis in Appendix 3.M.1 of the HI-STAR 100 FSAR~~ *the results presented in Subsection 3.4.4.3.1.8* demonstrates that the sheathing, which affixes the ~~Boral~~ *fixed neutron absorber* panel, remains in place during all credible accident conditions, and thus, the ~~Boral~~ *fixed neutron absorber* panel remains permanently fixed. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

### 6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 1 of this FSAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel positioning was judged to be not significant. Starting in Revision 2 of this FSAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for all new or changed conditions. The calculations in this subsection serve to determine for which of these conditions the eccentric positioning of assemblies in the fuel storage locations results in a higher maximum  $k_{\text{eff}}$  value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum  $k_{\text{eff}}$  value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4. All other calculations throughout this chapter, such as studies to determine bounding fuel dimensions, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- *Cell Center Configuration:* All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;
- *Basket Center Configuration:* All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and
- *Basket Periphery Configuration:* All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be  $(1/5)^{24}$  or approximately  $10^{-17}$  for the MPC-24,  $(1/5)^{32}$  or approximately  $10^{-23}$  for the MPC-32, and  $(1/5)^{68}$  or approximately  $10^{-48}$  for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

*In Table 6.3.5, results are presented for all conditions that were introduced in Revision 2 of this FSAR, namely results for the MPC-24E/EF with intact and damaged fuel at 5 wt% <sup>235</sup>U, for the MPC-32 with soluble boron levels lower than 2600 ppm for 5 wt% <sup>235</sup>U and lower than 1900 ppm for 4.1 wt% <sup>235</sup>U, and for the MPC-32 with intact and damaged fuel. The table shows the maximum  $k_{eff}$  value for centered and the two eccentric configurations for each condition, and the difference in  $k_{eff}$  between the centered and eccentric positioning. The results and conclusions are summarized as follows:*

- In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.*
- For the MPC-24E/EF, moving the assemblies and DFCs towards the center of the basket also results in a minor reduction. The cell centered configuration is therefore bounding for this condition and is used in the design basis calculations reported in Section 6.1 and Section 6.4.*
- For the MPC-32 cases listed in Table 6.3.5, the maximum reactivity is shown for the basket center configuration. However, for some of the cases with intact and damaged fuel in the MPC-32, the cell centered configuration results in a higher maximum reactivity. Therefore, both the cell centered and basket centered configuration are analyzed for the MPC-32 design basis calculation, and the higher results are listed in the tables in Section 6.1. and 6.4. This applies to the cases with intact and damaged fuel, and to cases with intact fuel only and soluble boron levels lower than 2600 ppm for 5 wt% <sup>235</sup>U and lower than 1900 ppm for 4.1 wt% <sup>235</sup>U.*

Table 6.3.1

## CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter <sup>†</sup>	$\Delta k$ for Maximum Tolerance		Action/Modeling Assumption
	MPC-24 <sup>‡</sup>	MPC-68	
Reduce <del>Boral</del> Fixed Neutron Absorber Width to Minimum	N/A <sup>†††</sup> min. = nom. = 7.5" and 6.25"	N/A <sup>†††</sup> min. = nom. = 4.75"	Assume minimum <del>Boral</del> fixed neutron absorber width
Increase UO <sub>2</sub> Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO <sub>2</sub> density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min. = 8.86" nom. = 8.92"	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" nom. = 8.92"	-0.0030 max. = 6.113" nom. = 6.053"	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" nom. = 1.15"	N/A	Assume minimum water gap in the MPC-24

<sup>†</sup> Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

<sup>‡</sup> Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

<sup>†††</sup> The ~~Boral~~fixed neutron absorber width for the MPC-68 is 4.75" +0.125", -0" , the ~~Boral~~fixed neutron absorber widths for the MPC-24 are 7.5" +0.125", -0" and 6.25" +0.125" -0" (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	$\Delta k$ Maximum Tolerance		Action/Modeling Assumption
	MPC-24 <sup>‡</sup>	MPC-68	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0030	-0.0039	
70°C	-0.0089	-0.0136	
100°C	-0.0162	-0.0193	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0251	-0.0241	
100°C	-0.0412	-0.0432	
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

<sup>‡</sup> Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES<sup>†</sup>

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated k <sub>eff</sub>
MPC-24 <sup>††</sup> (17x17A01 @ 4.0% Enrichment)						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9325±0.0008 <sup>†††</sup>
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9300±0.0008
nominal	(10.906")	nom. - 0.04"	(8.88")	nom. + 0.05"	(0.3625")	0.9305±0.0007
MPC-68 (8x8C04 @ 4.2% Enrichment)						
minimum	(6.43")	minimum	(5.993")	nominal	(1/4")	0.9307±0.0007
nominal	(6.49")	nominal	(6.053")	nominal	(1/4")	0.9274±0.0007
maximum	(6.55")	maximum	(6.113")	nominal	(1/4")	0.9272±0.0008
nom. + 0.05"	(6.54")	nominal	(6.053")	nom. + 0.05"	(0.30")	0.9267±0.0007

Notes:

1. Values in parentheses are the actual value used.

<sup>†</sup> Tolerance for pitch and box I.D. are ± 0.06".  
Tolerance for box wall thickness is +0.05", -0.00".

<sup>††</sup> All calculations for the MPC-24 assume minimum water gap thickness (1.09").

<sup>†††</sup> Numbers are 1σ statistical uncertainties.



Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES<sup>†</sup>

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 <sup>††</sup>
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 ppm soluble boron						
minimum	(9.158")	minimum	(8.69")	nominal	(9/32")	0.9085±0.0007
nominal	(9.218")	nominal	(8.75")	nominal	(9/32")	0.9028±0.0007
maximum	(9.278")	maximum	(8.81")	nominal	(9/32")	0.8996±0.0008
nominal+0.05"	(9.268")	nominal	(8.75")	nominal+0.05"	(0.331")	0.9023±0.0008
minimum+0.05"	(9.208")	minimum	(8.69")	nominal+0.05"	(0.331")	0.9065±0.0007
maximum	(9.278")	Maximum-0.05"	(8.76")	nominal+0.05"	(0.331")	0.9030±0.0008

Notes:

1. Values in parentheses are the actual value used.

<sup>†</sup> Tolerance for pitch and box I.D. are ± 0.06".  
Tolerance for box wall thickness is +0.05", -0.00".

<sup>††</sup> Numbers are 1σ statistical uncertainties.

Table 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

<b>Basket Type</b>	<b>Pitch</b>	<b>Box I.D.</b>	<b>Box Wall Thickness</b>	<b>Water-Gap Flux Trap</b>
MPC-24	nominal (10.906")	maximum (8.98")	nominal (5/16")	minimum (1.09")
MPC-24E	nominal (10.847")	maximum (8.81", 9.11" for DFC Positions)	nominal (5/16")	minimum (1.076", 0.776" for DFC Positions)
MPC-32	Minimum (9.158")	Minimum (8.69")	Nominal (9/32")	N/A
MPC-68	minimum (6.43")	Minimum (5.993")	nominal (1/4")	N/A

Table 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>MPC-24, MPC-24E and MPC-32</b>		
<b>UO<sub>2</sub> 5.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.696E-02	1.185E-01
92235	1.188E-03	4.408E-02
92238	2.229E-02	8.374E-01
<b>UO<sub>2</sub> 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
<b>BORAL (0.02 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
<b>BORAL (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24E and MPC-32)</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>METAMIC (0.02 g<sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.648 (MPC-24)</b>		
<i>Nuclide</i>	<i>Atom-Density</i>	<i>Wgt. Fraction</i>
5010	6.314E-03	3.965E-02
5011	2.542E-02	1.755E-01
6012	7.932E-02	5.975E-02
13027	4.286E-02	7.251E-01
<b>METAMIC (0.0279 g<sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.646 (MPC-24E and MPC-32)</b>		
<i>Nuclide</i>	<i>Atom-Density</i>	<i>Wgt. Fraction</i>
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>BORATED WATER, 300 PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wt. Fraction</b>
5010	3.248E-06	5.400E-05
5011	1.346E-05	2.460E-04
1001	6.684E-02	1.1186E-01
8016	3.342E-02	8.8784E-01
<b>BORATED WATER, 400PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	4.330E-06	7.200E-05
5011	1.794E-05	3.280E-04
1001	6.683E-02	1.1185E-01
8016	3.341E-02	8.8775E-01
<b>BORATED WATER, 1900PPM, DENSITY (g/cc)=1.00</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	2.057E-05	3.420E-04
5011	8.522E-05	1.558E-03
1001	6.673E-02	1.1169E-01
8016	3.336E-02	8.8641E-01
<b>BORATED WATER, 2600PPM, DENSITY (g/cc)=0.93</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	2.618e-05	4.680E-04
5011	1.085e-04	2.132E-03
1001	6.201e-02	1.1161E-01
8016	3.101e-02	8.8579E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>MPC-68</b>		
<b>UO<sub>2</sub> 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
<b>UO<sub>2</sub> 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
<b>MOX FUEL<sup>†</sup>, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

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<sup>†</sup> The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>BORAL (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.660</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
<b>METAMIC (0.0279 g <sup>10</sup>B/cm sq), DENSITY (g/cc) = 2.646</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01
<b>FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
50408016	4.798E-02	1.212E-01
504492235	4.001E-04	1.484E-02
601292238	2.742E-05	1.030E-03
1302790232	2.357E-02	8.630E-01
<b>COMMON MATERIALS</b>		
<b>ZR CLAD, DENSITY (g/cc) = 6.550</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
40000	4.323E-02	1.000E+00
<b>MODERATOR (H<sub>2</sub>O), DENSITY (g/cc) = 1.000</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>STAINLESS STEEL, DENSITY (g/cc) = 7.840</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
<b>ALUMINUM, DENSITY (g/cc) = 2.700</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
13027	6.026E-02	1.000E+00



Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

<b>CONCRETE, DENSITY (g/cc) = 2.35</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
1001	8.806E-03	6.000E-03
8016	4.623E-02	5.000E-01
11000	1.094E-03	1.700E-02
13027	2.629E-04	4.800E-03
14000	1.659E-02	3.150E-01
19000	7.184E-04	1.900E-02
20000	3.063E-03	8.300E-02
26000	3.176E-04	1.200E-02
<b>LEAD, DENSITY (g/cc) = 11.34</b>		
<b>Nuclide</b>	<b>Atom-Density</b>	<b>Wgt. Fraction</b>
82000	3.296E-02	1.0
<b>HOLTITE-A, DENSITY (g/cc) = 1.61</b>		
1001	5.695E-02	5.920E-02
5010	1.365E-04	1.410E-03
5011	5.654E-04	6.420E-03
6012	2.233E-02	2.766E-01
7014	1.370E-03	1.980E-02
8016	2.568E-02	4.237E-01
13027	7.648E-03	2.129E-01

Table 6.3.5

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT  
(FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

<i>CASE</i>	<i>Contents centered (Reference)</i>	<i>Content moved towards center of basket</i>		<i>Content moved towards basket periphery</i>	
	<i>Maximum <math>k_{eff}</math></i>	<i>Maximum <math>k_{eff}</math></i>	<i><math>k_{eff}</math> Difference to Reference</i>	<i>Maximum <math>k_{eff}</math></i>	<i><math>k_{eff}</math> Difference to Reference</i>
<i>MPC-24E/EF, Intact Fuel and Damaged Fuel/Fuel Debris, 5% Enrichment, 600ppm Soluble Boron</i>	0.9185	0.9178	-0.0007	0.9132	-0.0053
<i>MPC-32/32F, Intact Fuel, Assembly Class 16x16A, 4.1% Enrichment, 1300ppm Soluble Boron</i>	0.9429	0.9468	0.0039	0.9068	-0.0361
<i>MPC-32/32F, Intact Fuel, Assembly Class 15x15B, 5.0% Enrichment, 2400ppm Soluble Boron</i>	0.9473	0.9493	0.0020	0.9306	-0.0167
<i>MPC-32/32F, Intact Fuel and Damaged Fuel/Fuel Debris, Assembly Class 15x15F (Intact), 5% Enrichment, 2900ppm Soluble Boron</i>	0.9378	0.9397	0.0019	0.9277	-0.0101

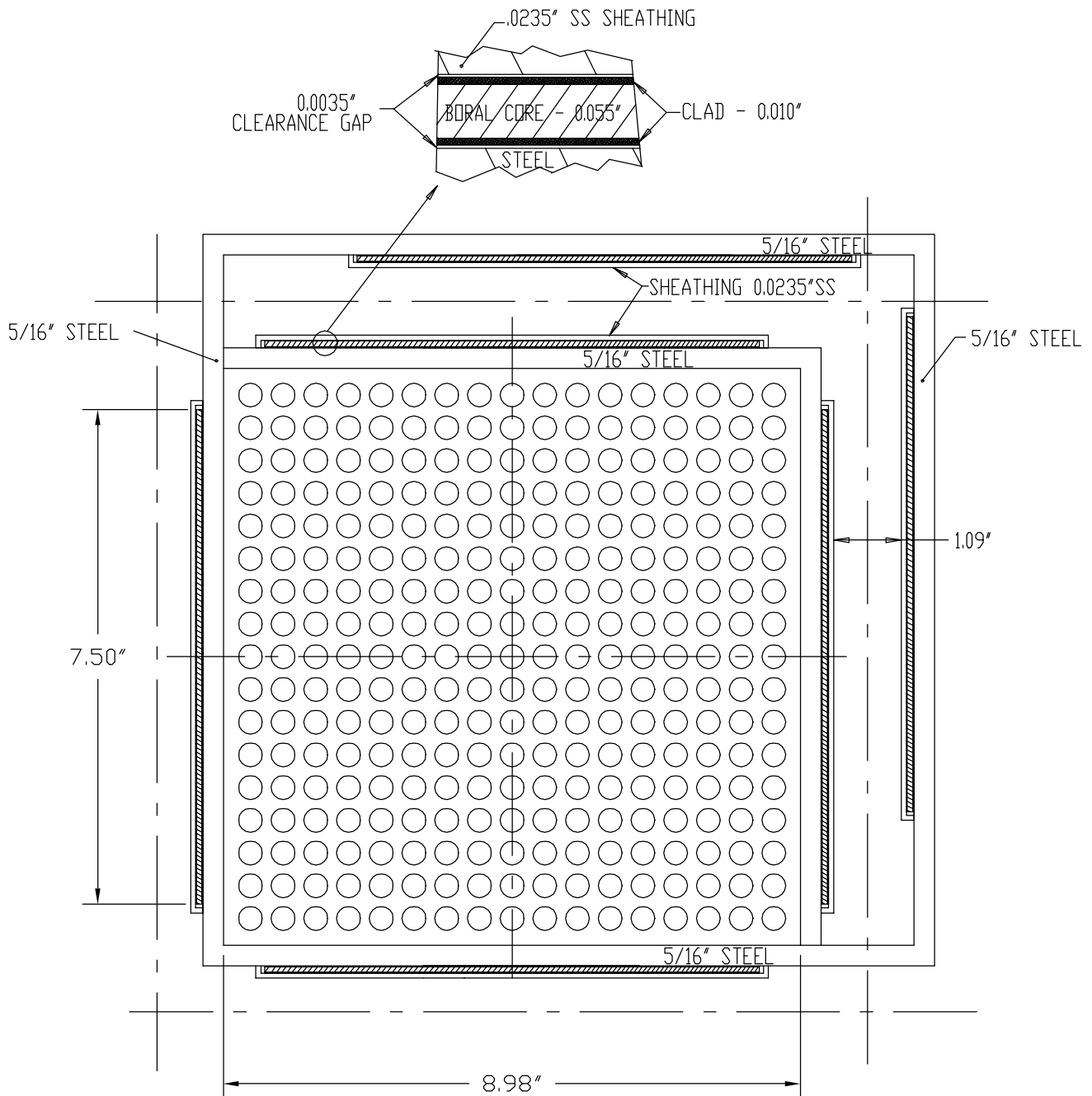


FIGURE 6.3.1; TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-24 BASKET  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

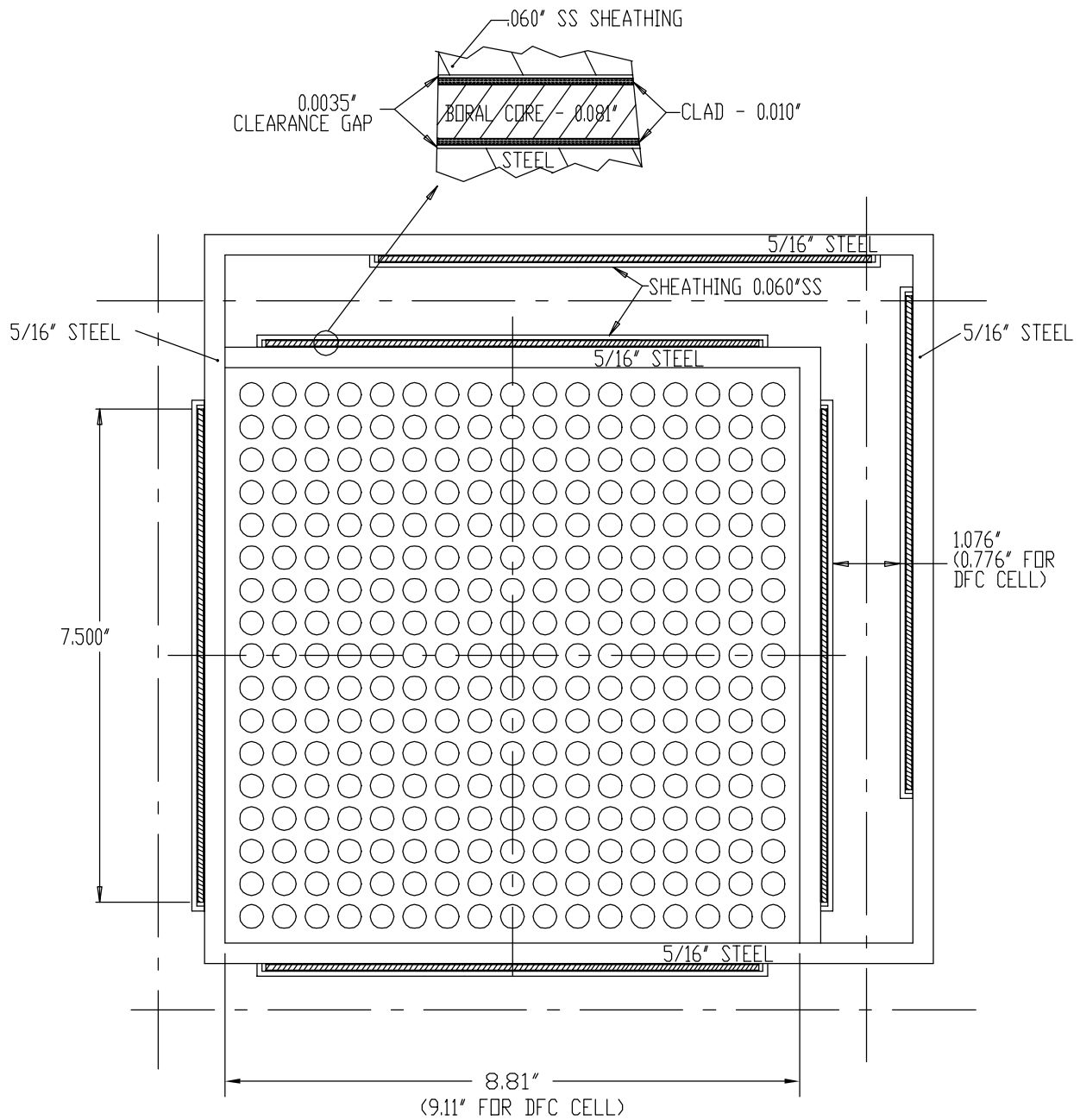


FIGURE 6.3.1A; TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-24E BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

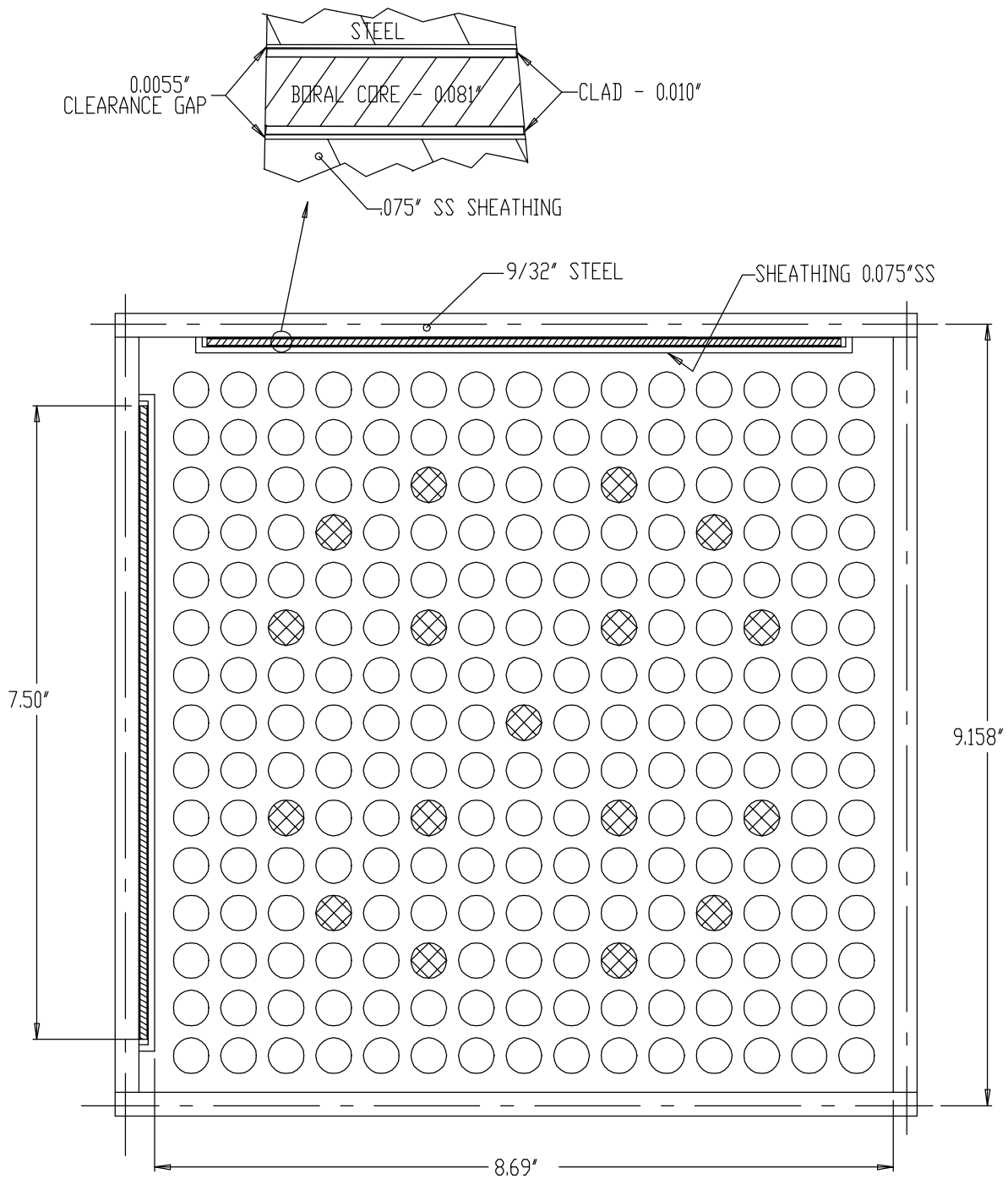


FIGURE 6.3.2; TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION)  
 WITH REPRESENTATIVE FUEL IN THE MPC-32 BASKET  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

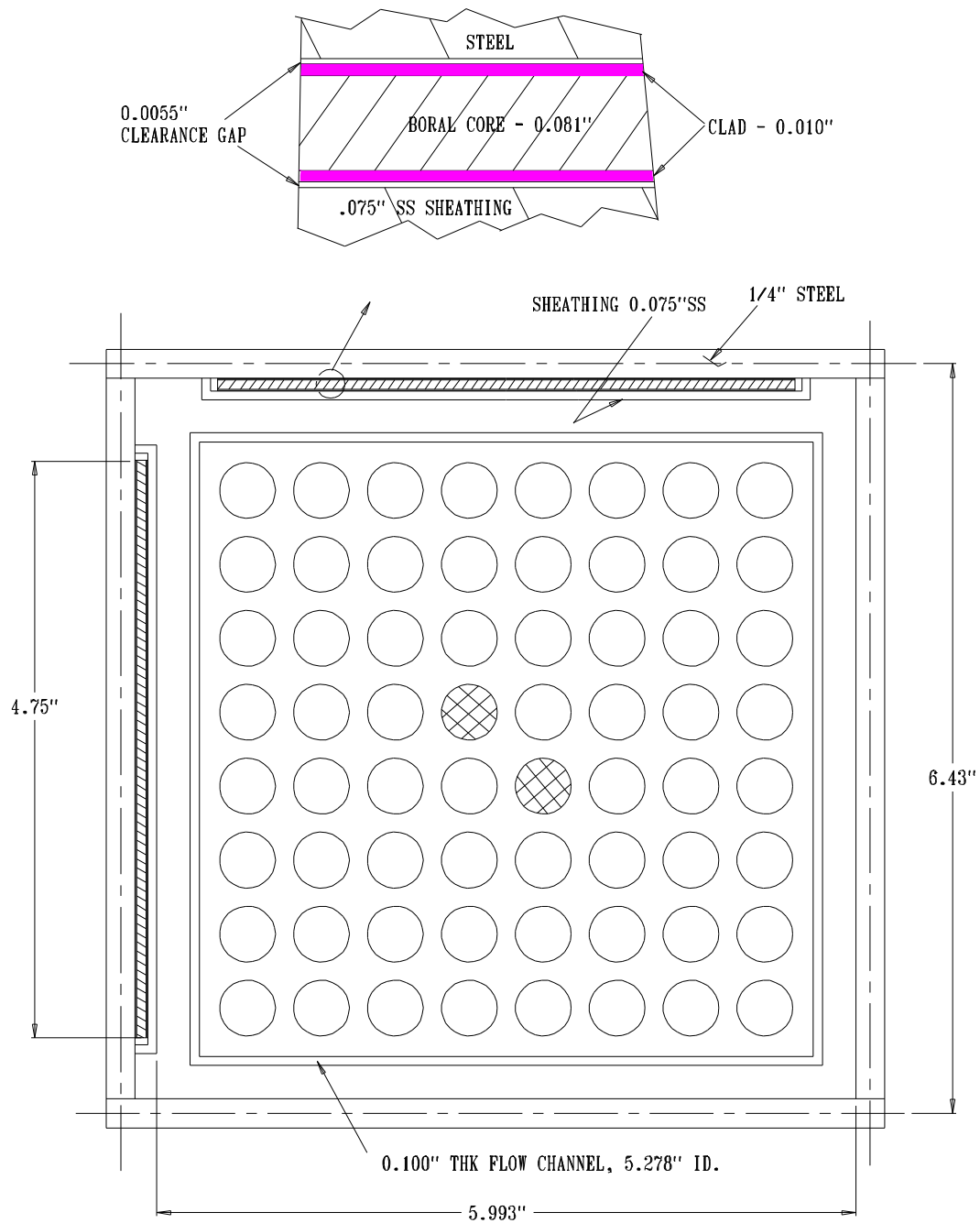


FIGURE 6.3.3; TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-68 BASKET ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

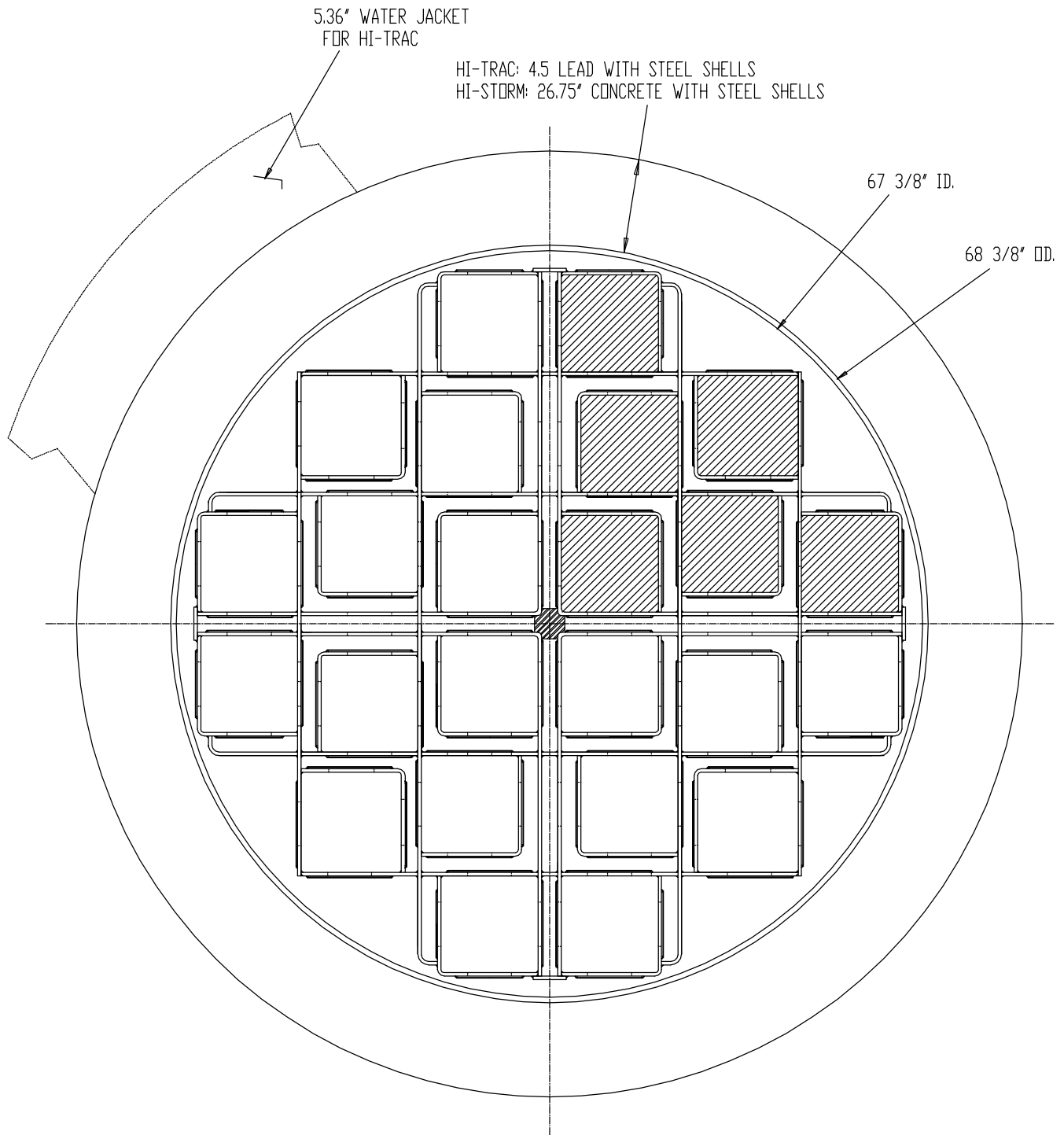


FIGURE 6.3.4; CALCULATION MODEL (PLANAR CROSS-SECTION)  
 WITH FUEL ILLUSTRATED IN ONE QUADRANT OF  
 THE MPC -24 AND THE MPC-24E.

( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

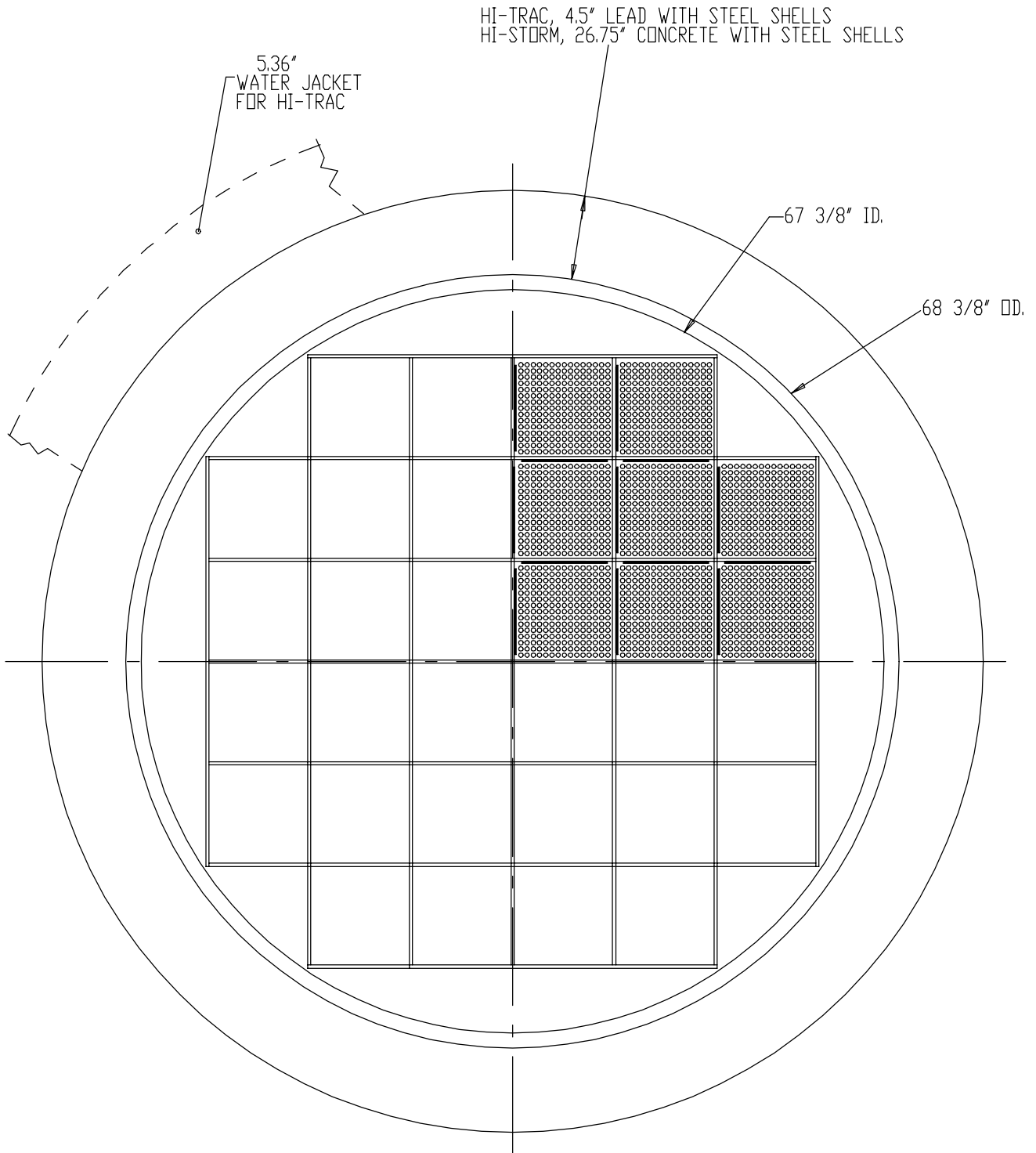


FIGURE 6.3.5; CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH FUEL ILLUSTRATED IN ONE QUADRANT OF  
THE MPC-32.

( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )



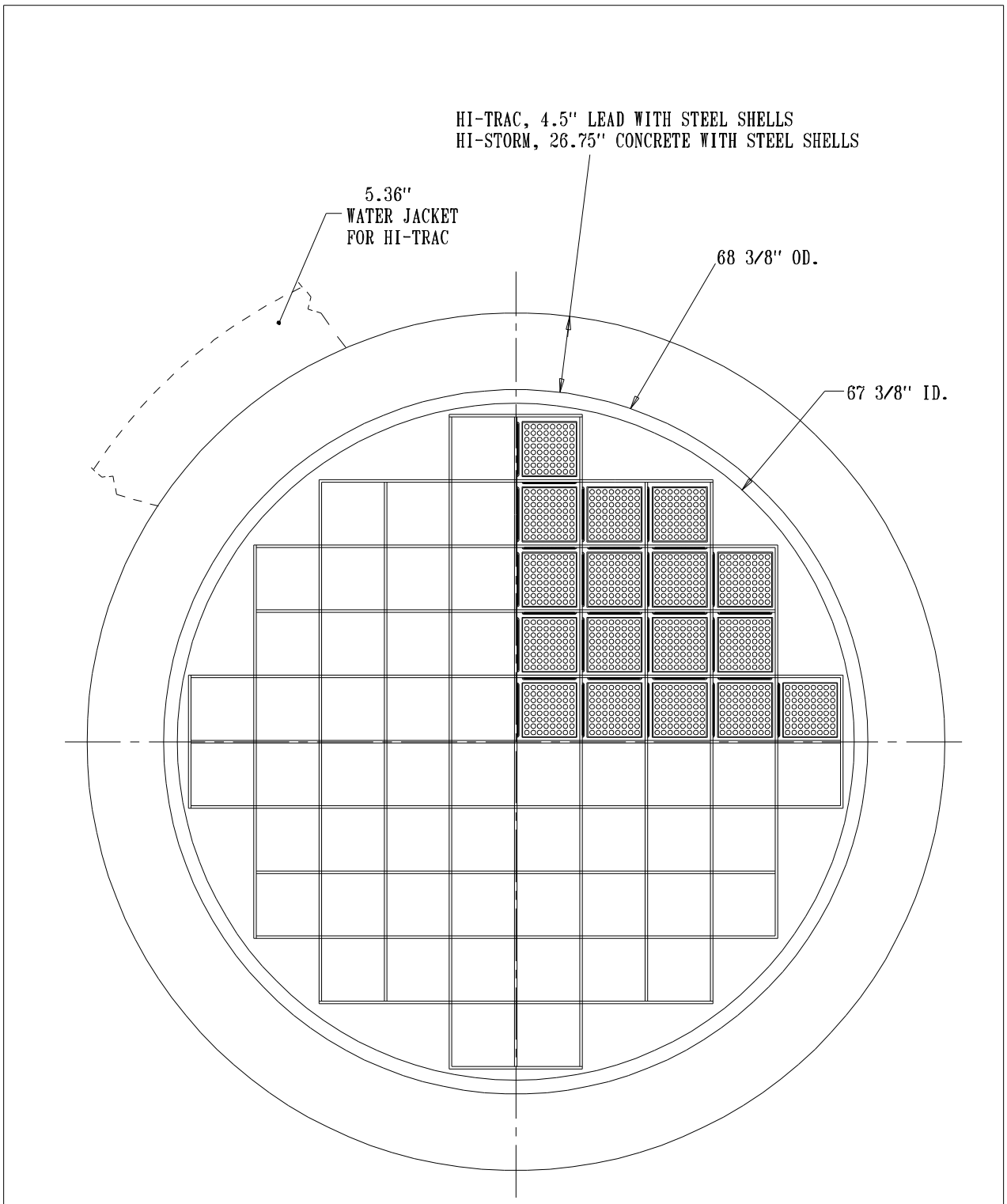


FIGURE 6.3.6; CALCULATION MODEL ( PLANAR CROSS-SECTION )  
 WITH FUEL ILLUSTRATED IN ONE QUADRANT OF  
 THE MPC-68  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

	ACTIVE FUEL LENGTH	LOWER WATER THICKNESS	UPPER WATER THICKNESS
MPC-68	SEE TABLE 6.2.1	7.30 IN.	8.46 IN.
MPC-24,-24E & -32	SEE TABLE 6.2.2	4.0 IN.	6.0 IN.

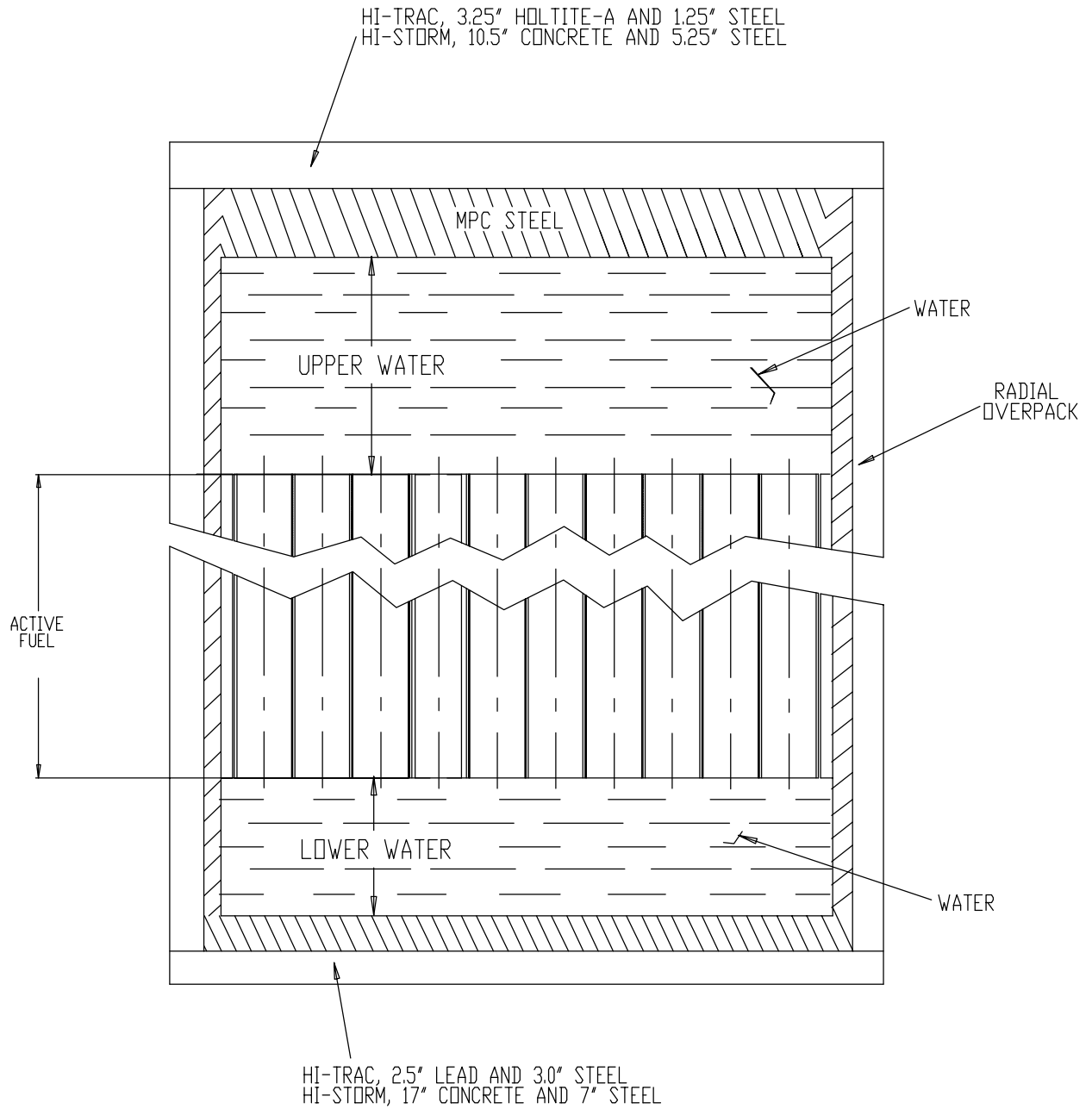


FIGURE 6.3.7; SKETCH OF THE CALCULATIONAL MODEL IN THE AXIAL DIRECTION

## 6.4 CRITICALITY CALCULATIONS

### 6.4.1 Calculational or Experimental Method

#### 6.4.1.1 Basic Criticality Safety Calculations

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, a minimum of 5,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket in the HI-STORM 100 System.

CASMO-3 [6.1.9] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO-3 has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO-3 calculations.

## 6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Tables 6.1.1 through 6.1.8 . These calculations were based on the assumption that the HI-STORM 100 System (HI-TRAC transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

~~Nominally, the fuel assemblies would be centrally positioned in each MPC basket cell. However, in accordance with NUREG-1536, the consequence of eccentric positioning was also evaluated and found to be negligible. To simulate eccentric positioning (and possible closer approach to the MPC steel shield), calculations were performed analytically decreasing the inner radius until it was 1 cm away<sup>†</sup> from the nearest fuel. Results showed a minor increase in reactivity of 0.0026  $\Delta k$  maximum (MPC-68) which implies that the effect of eccentric location of fuel is negligible at the actual reflector spacing.~~

### 6.4.2.1 Internal and External Moderation

As required by NUREG-1536, calculations in this section demonstrate that the HI-STORM 100 System remains subcritical for all credible conditions of moderation.

#### 6.4.2.1.1 Unborated Water

With a neutron absorber present (i.e., the ~~Bora~~*fixed neutron absorber* sheets or the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask were made to confirm that the phenomenon does not occur with low density water inside or outside the casks.

Calculations for the MPC designs with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case

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<sup>†</sup> PNL critical experiments have shown a small positive reactivity effect of thick steel reflectors, with the maximum effect at 1 cm distance from the fuel. In the cask designs, the fuel is mechanically prohibited from being positioned at a 1 cm spacing from the overpack steel.

1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Figure 6.4.10 plots calculated  $k_{\text{eff}}$  values ( $\pm 2\sigma$ ) as a function of internal moderator density for both MPC designs with 100% external moderator density (i.e., full water reflection). Results listed in Table 6.4.1 support the following conclusions:

- For each type of MPC, the calculated  $k_{\text{eff}}$  for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and
- For each type of MPC, reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STORM 100 System.

For each of the MPC designs, the maximum  $k_{\text{eff}}$  values are shown to be less than or statistically equal to that of a single internally flooded unreflected cask and are below the regulatory limit of 0.95.

#### 6.4.2.1.2 Borated Water

With the presence of a soluble neutron absorber in the water, the discussion in the previous section is not always applicable. Calculations were made to determine the optimum moderator density for the MPC designs that require a minimum soluble boron concentration.

Calculations for the MPC designs with various internal moderator densities are shown in Table 6.4.6. As shown in the previous section, the external moderator density has a negligible effect on the reactivity, and is therefore not varied. Water containing soluble boron has a slightly higher density than pure water. Therefore, water densities up to 1.005 g/cm<sup>3</sup> were analyzed for the higher soluble boron concentrations. Additionally, for the higher soluble boron concentrations, analysis have been performed with empty (voided) guide tubes. This variation is discussed in detail in Section 6.4.8. Results listed in the Table 6.4.6 support the following conclusions:

- For all cases with a soluble boron concentration of up to 1900ppm, and for a soluble boron concentration of 2600ppm assuming voided guide tubes, the conclusion of the Section 6.4.2.1.1 applies, i.e. the maximum reactivity is corresponds to 100% moderator density.
- For 2600ppm soluble boron concentration with filled guide tubes, the results presented in Table 6.4.6 indicate that there is a maximum of the reactivity somewhere between 0.90 g/cm<sup>3</sup> and 1.00 g/cm<sup>3</sup> moderator density. However, a distinct maximum can not be

identified, as the reactivities in this range are very close. For the purpose of the calculations with 2600ppm soluble boron concentration, a moderator density of 0.93 g/cm<sup>3</sup> was chosen, which corresponds to the highest calculated reactivity listed in Table 6.4.6.

*The calculations documented in this chapter also use soluble boron concentrations other than 1900 ppm and 2600 ppm in the MPC-32/32F. For the MPC-32 loaded with intact fuel only, soluble boron concentrations between 1300 ppm and 2600 ppm are used. For the MPC-32/32F loaded with intact fuel, damaged fuel and fuel debris, soluble boron concentrations between 1500 ppm and 2900 ppm are used. In order to determine the optimum moderation condition for each assembly class at the corresponding soluble boron level, evaluations are performed with filled and voided guide tubes, and for water densities of 1.0 g/cm<sup>3</sup> and 0.93 g/cm<sup>3</sup> for each class and enrichment level. Results for the MPC-32 loaded with intact fuel only are listed in Table 6.4.10 for an initial enrichment of 5.0 wt% <sup>235</sup>U and in Table 6.4.11 for an initial enrichment of 4.1 wt% <sup>235</sup>U. Corresponding results for the MPC-32/32F loaded with intact fuel, damaged fuel and fuel debris are listed in Table 6.4.14. The highest value listed in these tables for each assembly class is listed as the bounding value in Section 6.1.*

#### 6.4.2.2 Partial Flooding

As required by NUREG-1536, calculations in this section address partial flooding in the HI-STORM 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

#### 6.4.2.3 Clad Gap Flooding

As required by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum  $k_{\text{eff}}$  values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

#### 6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e. different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 11). Because the fuel cladding temperatures remain below their design limits (as demonstrated in Chapter 4) and the inertial loading remains below 63g's (the inertial loadings associated with the design basis drop accidents discussed in Chapter 11 are limited to 45g's), the cladding remains intact (see Section 3.5). For damaged fuel assemblies and fuel debris, the assemblies or debris are pre-loaded into stainless steel Damaged Fuel Containers fitted with 250x250 fine mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. This condition would be the result of the water tension across the mesh screens. The maximum water level inside the DFCs for this condition is calculated from the dimensions of the mesh screen and the surface tension of water. The wetted perimeter of the screen openings is 50 ft per square inch of screen. With a surface tension of water of 0.005 lbf/ft, this results in a maximum pressure across the screen of 0.25 psi, corresponding to a maximum water height in the DFC of 7 inches. For added conservatism, a value of 12 inches is used. Assuming this condition, calculations are performed for all three possible DFC configurations:

- MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)
- MPC-68 or MPC-68FF with 16 DFCs (All BWR Assembly Classes)
- MPC-24E or MPC-24EF with 4 DFCs (All PWR Assembly Classes)
- *MPC-32 or MPC-32F with 8 DFCs (All PWR Assembly Classes)*

For each configuration, the case resulting in the highest maximum  $k_{\text{eff}}$  for the fully flooded condition (see Section 6.4.4) is re-analyzed assuming the preferential flooding condition. For these analyses, the lower 12 inches of the active fuel in the DFCs and the water region below the active fuel (see Figure 6.3.7) are filled with full density water (1.0 g/cc). The remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). Table 6.4.4 lists the maximum  $k_{\text{eff}}$  for the ~~three~~four configurations in comparison with the maximum  $k_{\text{eff}}$  for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum  $k_{\text{eff}}$  than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC confinement boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Section 6.4.4.

#### 6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 11 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC transfer cask and minor damage to the concrete radiation shield for the HI-STORM storage cask, which have no adverse effect on the design parameters important to criticality safety.

As reported in Chapter 3, Table 3.4.4, the minimum factor of safety for either MPC as a result of the hypothetical cask drop or tip-over accident is 1.1 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change will be insignificant compared to the characteristic dimension of the flux trap.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM 100 System is in full compliance with the requirement of 10CRF72.124, which states that “before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.”

#### 6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) are presented in section 6.2 and summarized in Section 6.1. To demonstrate the applicability of the HI-STAR analyses, results of the design basis criticality safety calculations for the HI-STAR cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor ( $k_{\text{eff}}$ ), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-3) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.



In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff}^{max} = k_c + K_c \sigma_c + Bias + \sigma_B$$

where:

- ⇒  $k_c$  is the calculated  $k_{eff}$  under the worst combination of tolerances;
- ⇒  $K_c$  is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final  $k_{eff}$  value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle  $k_{eff}$  values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;
- ⇒  $\sigma_c$  is the standard deviation of the calculated  $k_{eff}$ , as determined by the computer code (MCNP4a or KENO5a);
- ⇒ **Bias** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and
- ⇒  $\sigma_B$  is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

#### 6.4.4 Damaged Fuel and Fuel Debris

Damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. ~~Four (4)~~ *Five (5)* different DFC types with different cross sections are analyzed. Three (3) of these DFCs are designed for BWR fuel assemblies, ~~one (1)~~ *two (2)* are designed for PWR fuel assemblies. Two of the DFCs for BWR fuel are specifically designed for fuel assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. These assemblies have a smaller cross section, a shorter active length and a low initial enrichment of 2.7 wt% <sup>235</sup>U, and therefore a low reactivity. The analysis for these assembly classes is presented in the following Section 6.4.4.1. The remaining ~~two (2)~~ *three (3)* DFCs are generic DFCs designed for all BWR and PWR assembly classes. The criticality analysis for these generic DFCs is presented in Section 6.4.4.2.

6.4.4.1 MPC-68, MPC-68F or MPC-68FF loaded with Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

This section only addresses criticality calculations and results for assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A, loaded into the MPC-68, MPC-68F or MPC-68FF. Up to 68 DFCs with these assembly classes are permissible to be loaded into the MPC. Two different DFC types with slightly different cross-sections are analyzed. DFCs containing fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing damaged fuel assemblies may be stored in either the MPC-68, MPC-68F or MPC-68FF. Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 and MPC-68FF have a higher specified  $^{10}\text{B}$  loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68 or MPC-68FF. Although the maximum planar-average enrichment of the damaged fuel is limited to 2.7%  $^{235}\text{U}$  as specified in the ~~Certificate of Compliance~~ *Section 2.1.9*, analyses have been made for three possible scenarios, conservatively assuming fuel<sup>††</sup> of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.2 through 6.4.8.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This configuration is illustrated in Figure 6.4.9.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.5, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in  $^{238}\text{U}$  neutron capture (higher effective resonance integral for  $^{238}\text{U}$  absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.5.

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<sup>††</sup> 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

The analyses performed and summarized in Table 6.4.5 provides the relative magnitude of the effects on the reactivity. This information coupled with the maximum  $k_{\text{eff}}$  values listed in Table 6.1.3 and the conservatism in the analyses, demonstrate that the maximum  $k_{\text{eff}}$  of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of  $k_{\text{eff}} < 0.95$ .

#### 6.4.4.2 Generic BWR and PWR Damaged Fuel and Fuel Debris

The MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68 and MPC-68FF are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into generic DFCs. The number of generic DFCs is limited to 16 for the MPC-68 and MPC-68FF, and to 4 for the MPC-24E and MPC-24EF, and to 8 for the MPC-32 and MPC-32F. The permissible locations of the DFCs are shown in Figure 6.4.11 for the MPC-68/68FF, and in Figure 6.4.12 for the MPC-24E/24EF and in Figure 6.4.16 for the MPC-32/32F.

Damaged fuel assemblies are assemblies with known or suspected cladding defects greater than pinholes or hairlines, or with missing rods, but excluding fuel assemblies with gross defects (for a full definition see Section 1.1 of the Certificate of Compliance Table 1.0.1). Therefore, apart from possible missing fuel rods, damaged fuel assemblies have the same geometric configuration as intact fuel assemblies and consequently the same reactivity. Missing fuel rods can result in a slight increase of reactivity. After a drop accident, however, it can not be assumed that the initial geometric integrity is still maintained. For a drop on either the top or bottom of the cask, the damaged fuel assemblies could collapse. This would result in a configuration with a reduced length, but increased amount of fuel per unit length. For a side drop, fuel rods could be compacted to one side of the DFC. In either case, a significant relocation of fuel within the DFC is possible, which creates a greater amount of fuel in some areas of the DFC, whereas the amount of fuel in other areas is reduced. Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

In the cases of fuel debris or relocated damaged fuel, there is the potential that fuel could be present in axial sections of the DFCs that are outside the basket height covered by ~~Boral~~ the fixed neutron absorber. However, in these sections, the DFCs are not surrounded by any intact fuel, only by basket cell walls, non-fuel hardware, water and for the MPC-68/68FF by a maximum of one other DFC. Studies have shown that this condition does not result in any significant effect on reactivity, compared to a condition where the damaged fuel and fuel debris is restricted to the axial section of the basket covered by ~~Boral~~ the fixed neutron absorber. All calculations for generic BWR and PWR damaged fuel and fuel debris are therefore performed assuming that fuel is present only in the axial sections covered by ~~Boral~~ the fixed neutron absorber, and the results are directly applicable to any situation where damaged fuel and fuel debris is located outside these sections in the DFCs.

To address all the situations listed above and identify the configuration or configurations leading to the highest reactivity, it is impractical to analyze a large number of different geometrical configurations for each of the fuel classes. Instead, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following sections.

All calculations for generic damaged fuel and fuel debris are performed using a full cask model with the maximum permissible number of Damaged Fuel Containers. For the MPC-68 and MPC-68FF, the model therefore contains 52 intact assemblies, and 16 DFCs in the locations shown in Figure 6.4.11. For the MPC-24E and MPC-24EF, the model consists of 20 intact assemblies, and 4 DFCs in the locations shown in Figure 6.4.12. *For the MPC-32 and MPC-32, the model consists of 24 intact assemblies, and 8 DFCs in the locations shown in Figure 6.4.16.* The bounding assumptions regarding the intact assemblies and the modeling of the damaged fuel and fuel debris in the DFCs are discussed in the following sections.

*Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term ‘damaged fuel’ as used throughout this chapter designates both damaged fuel and fuel debris.*

#### 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%  $^{235}\text{U}$ , 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%  $^{235}\text{U}$ , 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%  $^{235}\text{U}$  *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%  $^{235}\text{U}$ , 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 shows 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 and G;*
- *15x15F for assembly classes 15x15D, E, F and H;*
- *16x16A; and*
- *17x17C for all 17x17 assembly classes.*

#### 6.4.4.2.2 Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e. all cladding and other structural material in the DFC is replaced by water.
- *For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.*
- The active length of these rods is chosen to be the maximum active fuel length of all fuel assemblies listed in Section 6.2, which is 155 inch for BWR fuel and 150 inch for PWR fuel.
- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 9 (3x3) and 189 (17x17) for BWR fuel, and between 64 (8x8) and 729 (27x27) for PWR fuel.
- Analyses are performed for the minimum, maximum and typical pellet diameter of PWR and BWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

This is also a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 155 inch (BWR) or 150 inch (PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

To demonstrate the level of conservatism, additional analyses are performed with the DFC containing various realistic assembly configurations such as intact assemblies, assemblies with missing fuel rods and collapsed assemblies, i.e. assemblies with increased number of rods and decreased rod pitch.

As discussed in Section 6.4.4.2, all calculations are performed for full cask models, containing the maximum permissible number of DFCs together with intact assemblies.

*As an example of the damaged fuel model used in the analyses, Figure 6.4.17 shows the basket cell of an MPC-32 with a DFC containing a 17x17 array of bare fuel rods.*

Graphical presentations of the calculated maximum  $k_{\text{eff}}$  for each typical cases as a function of the fuel mass per unit length of the DFC are shown in Figures 6.4.13 (BWR) and 6.4.14 (PWR, MPC-24E/EF with pure water). The results for the bare fuel rods show a distinct peak in the maximum  $k_{\text{eff}}$  at about 2 kg UO<sub>2</sub>/inch for BWR fuel, and at about 3.5 kgUO<sub>2</sub>/inch for PWR fuel.

The realistic assembly configurations are typically about 0.01 (delta-k) or more below the peak results for the bare fuel rods, demonstrating the conservatism of this approach to model damaged fuel and fuel debris configurations such as severely damaged assemblies and bundles of fuel rods.

For fuel debris configurations consisting of bare fuel pellets only, the fuel mass per unit length would be beyond the value corresponding to the peak reactivity. For example, for DFCs filled with a mixture of 60 vol% fuel and 40 vol% water the fuel mass per unit length is 3.36 kgUO<sub>2</sub>/inch for the BWR DFC and 7.92 kgUO<sub>2</sub>/inch for the PWR DFC. The corresponding reactivities are significantly below the peak reactivities. The difference is about 0.005 (delta-k) for BWR fuel and 0.01 (delta-k) or more for PWR fuel. Furthermore, the filling height of the DFC would be less than 70 inches in these examples due to the limitation of the fuel mass per basket position, whereas the calculation is conservatively performed for a height of 155 inch (BWR) or 150 inch (PWR). These results demonstrate that even for the fuel debris configuration of bare fuel pellets, the model using bare fuel rods is a conservative approach.

#### 6.4.4.2.3 Distributed Enrichment in BWR Fuel

BWR fuel usually has an enrichment distribution in each planar cross section, and is characterized by the maximum planar average enrichment. For intact fuel it has been shown that

using the average enrichment for each fuel rod in a cross section is conservative, i.e. the reactivity is higher than calculated for the actual enrichment distribution (See Appendix 6.B). For damaged fuel assemblies, additional configurations are analyzed to demonstrate that the distributed enrichment does not have a significant impact on the reactivity of the damaged assembly under accident conditions. Specifically, the following two scenarios were analyzed:

- As a result of an accident, fuel rods with lower enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the top part, but at the same time the amount of fuel in that area is reduced compared to the intact assembly.
- As a result of an accident, fuel rods with higher enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the bottom part, and at the same time the amount of fuel in that area is increased compared to the intact assembly, leading to a reduction of the water content.

In both scenarios, a compensation of effects on reactivity is possible, as the increase of reactivity due to the increased planar average enrichment might be offset by the possible reduction of reactivity due to the change in the fuel to water ratio. A selected number of calculations have been performed for these scenarios and the results show that there is only a minor change in reactivity. These calculations are shown in Figure 6.4.13 in the group of the explicit assemblies. Consequently, it is appropriate to qualify damaged BWR fuel assemblies and fuel debris based on the maximum planar average enrichment. For assemblies with missing fuel rods, this maximum planar average enrichment has to be determined based on the enrichment and number of rods still present in the assembly when loaded into the DFC.

#### 6.4.4.2.4 Results for MPC-68 and MPC-68FF

The MPC-68 and MPC-68FF allows the storage of up to sixteen DFCs in the shaded cells on the periphery of the basket shown in Figure 6.4.11. In the MPC-68FF, up to 8 of these cells may contain DFCs with fuel debris. The various configurations outlined in Sections 6.4.4.2.2 and 6.4.4.2.3 are analyzed with an enrichment of the intact fuel of 3.7%  $^{235}\text{U}$  and an enrichment of damaged fuel or fuel debris of 4.0%  $^{235}\text{U}$ . For the intact assembly, the bounding assembly of the 9x9E and 9x9F fuel classes was chosen. This assembly has the highest reactivity of all BWR assembly classes for the initial enrichment of 3.7 wt%  $^{235}\text{U}$ , as demonstrated in Table 6.4.7. The results for the various configurations are summarized in Figure 6.4.13 and in Table 6.4.8. Figure 6.4.13 shows the maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, for various actual and hypothetical damaged fuel or fuel debris configurations as a function of the fuel mass per unit length of the DFC. Table 6.4.8 lists the highest maximum  $k_{\text{eff}}$  for the various configurations. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.

#### 6.4.4.2.5 Results for MPC-24E and MPC-24EF

The MPC-24E allows the storage of up to four DFCs with damaged fuel in the four outer fuel basket cells shaded in Figure 6.4.12. The MPC-24EF allows storage of up to four DFCs with damaged fuel or fuel debris in these locations. These locations are designed with a larger box ID to accommodate the DFCs. For an enrichment of 4.0 wt%  $^{235}\text{U}$  for the intact fuel, damaged fuel and fuel debris, *and assuming no soluble boron*, the results for the various configurations outlined in Section 6.4.4.2.2 are summarized in Figure 6.4.14 and in Table 6.4.9. Figure 6.4.14 shows the maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, for various actual and hypothetical damaged fuel and fuel debris configurations as a function of the fuel mass per unit length of the DFC. For the intact assemblies, the 15x15H assembly class was chosen. This assembly class has the highest reactivity of all PWR assembly classes for a given initial enrichment. This is demonstrated in Table 6.1.4. Table 6.4.9 lists the highest maximum  $k_{\text{eff}}$  for the various configurations. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.

*For an enrichment of 5.0 wt%  $^{235}\text{U}$  for the intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration of 600 ppm is required. For this condition, calculations are performed for various hypothetical fuel debris configurations (i.e. bare fuel rods) as a function of the fuel mass per unit length of the DFC. Additionally, calculations are performed with reduced water densities in the DFC. The various conditions of damaged fuel, such as assemblies with missing rods or collapsed assemblies, were not analyzed, 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. Again, the 15x15H assembly class was chosen as the intact assembly since this assembly class has the highest reactivity of all PWR assembly classes as demonstrated in Table 6.1.4. The results are summarized in Table 6.4.12. Similar to the calculations with pure water (see Figure 6.4.14), the results for borated water show a distinct peak of the maximum  $k_{\text{eff}}$  as a function of the fuel mass per unit length. Therefore, for each condition, the table lists only the highest maximum  $k_{\text{eff}}$ , including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the reactivity decreases with decreasing water density. This demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. All maximum  $k_{\text{eff}}$  values are below the 0.95 regulatory limit.*

#### 6.4.4.2.6 Results for MPC-32 and MPC-32F

*The MPC-32 allows the storage of up to eight DFCs with damaged fuel in the outer fuel basket cells shaded in Figure 6.4.16. The MPC-32F allows storage of up to eight DFCs with damaged fuel or fuel debris in these locations. For the MPC-32 and MPC-32F, additional cases are analyzed due to the high soluble boron level required for this basket:*

- The assembly classes of the intact assemblies are grouped, and minimum required soluble boron levels are determined separately for each group. The analyses are*



performed for the bounding assembly class in each group. The bounding assembly classes are listed in Section 6.4.4.2.1.

- Evaluations of conditions with voided and filled guide tubes and various water densities in the MPC and DFC are performed to identify the most reactive condition.

*In general, all calculations performed for the MPC-32 show the same principal behavior as for the MPC-24 (see Figure 6.4.14), i.e. the reactivity as a function of the fuel mass per unit length for the bare fuel rod array shows a distinct peak. Therefore, for each condition analyzed, only the highest maximum  $k_{eff}$ , i.e. the calculated peak reactivity, is listed in the tables. Evaluations of different diameters of the bare fuel pellets and the reduced water density in the DFC have been performed for a representative case using the 15x15F assembly class as the intact assembly, with 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%  $^{235}\text{U}$  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%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , 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<sup>3</sup> in the MPC. In all cases, the water density inside the DFCs is assumed to be 1.0 g/cm<sup>3</sup>, 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.

#### 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  $k_{\text{eff}}$  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  $^{235}\text{U}$  (equivalent to  $\text{UO}_2$  fuel with an enrichment of approximately 1.7 wt%  $^{235}\text{U}$ ), 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  $k_{\text{eff}}$  values listed in Tables 6.1.7 and 6.1.8 this result demonstrates, that the  $k_{\text{eff}}$  for a Thoria Rod Canister loaded into the MPC-68 or the MPC-68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of  $k_{\text{eff}} < 0.95$ .

#### 6.4.7 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

#### 6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware such as Thimble Plugs (TPs), Burnable Poison Rod Assemblies (BPRAs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and similar devices are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged. This conclusion is supported by the calculation listed in Table 6.2.4, which shows a significant reduction in reactivity as a result of voided guide tubes, i.e. the removal of the water from the guide tubes.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding

approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes are filled or assumed empty. For the MPC-24 with 400 ppm and the MPC-32 with 1900 ppm, voiding the guide tubes results in a reduction of reactivity. All calculations for the MPC-24 and MPC-24E, and for the MPC-32 with 1900 ppm are therefore performed with water in the guide tubes. For the MPC-32 with 2600 ppm, the reactivity for voided guide tubes slightly exceeds the reactivity for filled guide tubes. However, this effect is not consistent across all assembly classes. Table 6.4.10, Table 6.4.11 and Table 6.4.14 shows results with filled and voided guide tubes for all assembly classes in the MPC-32/32F at 2600 ppm 4.1 wt% <sup>235</sup>U and 5.0 wt% <sup>235</sup>U. Some classes show an increase, other classes show a decrease as a result of voiding the guide tubes. Therefore, for the results presented in the Section 6.1, Table 6.1.5, Table 6.1.6 and Table 6.1.12, the maximum value for each class is chosen for each enrichment level.

In summary, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

#### 6.4.9 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a keff less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

#### 6.4.10 Applicability of HI-STAR Analyses to HI-STORM 100 System

Calculations previously supplied to the NRC in applications for the HI-STAR 100 System (Docket Numbers 71-9261 and 72-1008) are directly applicable to the HI-STORM storage and HI-TRAC transfer casks. The MPC designs are identical. The cask systems differ only in the overpack shield material. The limiting condition for the HI-STORM 100 System is the fully flooded HI-TRAC transfer cask. As demonstrated by the comparative calculations presented in Tables 6.1.1 through 6.1.8, the shield material in the overpack (steel and lead for HI-TRAC, steel

for HI-STAR) has a negligible impact on the eigenvalue of the cask systems. As a result, this analysis for the 125-ton HI-TRAC transfer cask is applicable to the 100-ton HI-TRAC transfer cask. In all cases, for the reference fuel assemblies, the maximum  $k_{\text{eff}}$  values are in good agreement and are conservatively less than the limiting  $k_{\text{eff}}$  value (0.95).

#### 6.4.11 Fixed Neutron Absorber Material

*The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorber materials: Boral or Metamic. Both materials are made of aluminum and  $B_4C$  powder. Boral has an inner core consisting of  $B_4C$  and aluminum between two outer layers consisting of aluminum only. This configuration is explicitly modeled in the criticality evaluation and shown in Figures 6.3.1 through 6.3.3 for each basket. Metamic is a single layer material with the same overall thickness and the same credited  $^{10}B$  loading (in  $g/cm^2$ ) for each basket. The majority of the criticality evaluations documented in this chapter are performed using Boral as the fixed neutron absorber. For a selected number of bounding cases, analyses are also performed using Metamic instead of Boral. The results for these cases are listed in Table 6.4.15, together with the corresponding result using Boral and the difference between the two materials for each case. Individual cases show small differences for the two materials. However, the differences are mostly below two times the standard deviation (the standard deviation is about 0.0008 for all cases in Table 6.4.15), indicating that the results are statistically equivalent. Furthermore, the average difference is well below one standard deviation, and all cases are below the regulatory limit of 0.95. In some cases listed in Table 6.4.15, the reactivity difference between Metamic and Boral might be larger than expected for two equivalent materials. Also, for four out of the five cases with MPC-24 type baskets, Metamic shows the higher reactivity, which could potentially indicate a trend rather than a statistical variation. Therefore, in order to confirm that the materials are equivalent, a second set of calculations was performed for Metamic, which was statistically independent from the set shown in Table 6.4.15. This was achieved by selecting a different starting value for the random number generator in the Monte Carlo calculations. The second set also shows some individual variations of the differences, and a low average difference. However, there is no apparent trend regarding the MPC-24 type baskets compared to the MPC-32 and MPC-68, and the maximum positive reactivity difference for Metamic in an MPC-24 type basket is only 0.0005. Overall, the calculations demonstrate that the two fixed neutron absorber materials are identical from a criticality perspective. All results obtained for Boral are therefore directly applicable to Metamic and no further evaluations using Metamic are required.*

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS<sup>†</sup>

Case Number	Water Density		MCNP4a Maximum $k_{\text{eff}}$ <sup>††</sup>	
	Internal	External	MPC-24 (17x17A01 @ 4.0%)	MPC-68 (8x8C04 @ 4.2%)
1	100%	single cask	0.9368	0.9348
2	100%	100%	0.9354	0.9339
3	100%	70%	0.9362	0.9339
4	100%	50%	0.9352	0.9347
5	100%	20%	0.9372	0.9338
6	100%	10%	0.9380	0.9336
7	100%	5%	0.9351	0.9333
8	100%	0%	0.9342	0.9338
9	70%	0%	0.8337	0.8488
10	50%	0%	0.7426	0.7631
11	20%	0%	0.5606	0.5797
12	10%	0%	0.4834	0.5139
13	5%	0%	0.4432	0.4763
14	10%	100%	0.4793	0.4946

<sup>†</sup> For an infinite square array of casks with 60cm spacing between cask surfaces.

<sup>††</sup> Maximum  $k_{\text{eff}}$  includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

## REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

<b>MPC-24 (17x17A01 @ 4.0% ENRICHMENT) (no soluble boron)</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
<b>MPC-68 (8x8C04 @ 4.2% ENRICHMENT)</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348
<b>MPC-32 (15x15F @ 5.0 % ENRICHMENT) 2600ppm Soluble Boron</b>			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.8927	31.25	0.9213
50	0.9215	50	0.9388
75	0.9350	68.75	0.9401
100	0.9445	100	0.9445

Notes:

1. All values are maximum  $k_{\text{eff}}$  which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLETT-TO-CLAD GAP

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded with unborated water	0.9368	0.9348

Notes:

1. All values are maximum  $k_{eff}$  which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.4

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Preferential Flooding	Fully Flooded
MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)	0.6560	0.7857
MPC-68 or MPC-68FF with 16 DFCs (All BWR Assembly Classes)	0.6646	0.9328
MPC-24E or MPC-24EF with 4 DFCs (All PWR Assembly Classes)	0.7895	0.9480
<i>MPC-32 or MPC-32 with 8 DFCs (All PWR Assembly Classes)</i>	<i>0.7213</i>	<i>0.9378</i>

Notes:

1. All values are maximum  $k_{\text{eff}}$  which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.



Table 6.4.5

MAXIMUM  $k_{eff}$  VALUES<sup>†</sup> IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Maximum <sup>††</sup> $k_{eff}$	
	DFC Dimensions: ID 4.93” THK. 0.12”	DFC Dimensions: ID 4.81” THK. 0.11”
<u>6x6 Fuel Assembly</u>		
6x6 Intact Fuel	0.7086	0.7016
w/32 Rods Standing	0.7183	0.7117
w/28 Rods Standing	0.7315	0.7241
w/24 Rods Standing	0.7086	0.7010
w/18 Rods Standing	0.6524	0.6453
Collapsed to 8x8 array	0.7845	0.7857
Dispersed Powder	0.7628	0.7440
<u>7x7 Fuel Assembly</u>		
7x7 Intact Fuel	0.7463	0.7393
w/41 Rods Standing	0.7529	0.7481
w/36 Rods Standing	0.7487	0.7444
w/25 Rods Standing	0.6718	0.6644

<sup>†</sup> These calculations were performed with a planar-average enrichment of 3.0% and a <sup>10</sup>B loading of 0.0067 g/cm<sup>2</sup>, which is 75% of a minimum <sup>10</sup>B loading of 0.0089 g/cm<sup>2</sup>. The minimum <sup>10</sup>B loading in the MPC-68F is 0.010 g/cm<sup>2</sup>. Therefore, the listed maximum  $k_{eff}$  values are conservative

<sup>††</sup> Maximum  $k_{eff}$  includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.6

MAXIMUM  $k_{\text{eff}}$  VALUES WITH REDUCED BORATED WATER DENSITIES

Internal Water Density <sup>†</sup> in g/cm <sup>3</sup>	Maximum $k_{\text{eff}}$				
	MPC-24 (400ppm) @ 5.0 %	MPC-32 (1900ppm) @ 4.1 %		MPC-32 (2600ppm) @ 5.0 %	
Guide Tubes	filled	filled	void	filled	void
1.005	NC <sup>††</sup>	0.9403	0.9395	NC	0.9481
1.00	0.9314	0.9411	0.9400	0.9445	0.9483
0.99	NC	0.9393	0.9396	0.9438	0.9462
0.98	0.9245	0.9403	0.9376	0.9447	0.9465
0.97	NC	0.9397	0.9391	0.9453	0.9476
0.96	NC	NC	NC	0.9446	0.9466
0.95	0.9186	0.9380	0.9384	0.9451	0.9468
0.94	NC	NC	NC	0.9445	0.9467
0.93	0.9130	0.9392	0.9352	0.9465	0.9460
0.92	NC	NC	NC	0.9458	0.9450
0.91	NC	NC	NC	0.9447	0.9452
0.90	0.9061	0.9384	NC	0.9449	0.9454
0.80	0.8774	0.9322	NC	0.9431	0.9390
0.70	0.8457	0.9190	NC	0.9339	0.9259
0.60	0.8095	0.8990	NC	0.9194	0.9058
0.40	0.7225	0.8280	NC	0.8575	0.8410
0.20	0.6131	0.7002	NC	0.7421	0.7271
0.10	0.5486	0.6178	NC	0.6662	0.6584

<sup>†</sup> External moderator is modeled at 0%. This is consistent with the results demonstrated in Table 6.4.1.

<sup>††</sup> NC: Not Calculated

Table 6.4.7

MAXIMUM  $k_{\text{eff}}$  VALUES FOR INTACT BWR FUEL ASSEMBLIES WITH A MAXIMUM PLANAR AVERAGE ENRICHMENT OF 3.7 wt%  $^{235}\text{U}$

Fuel Assembly Class	Maximum $k_{\text{eff}}$
6x6A	0.8287
6x6C	0.8436
7x7A	0.8399
7x7B	0.9109
8x8A	0.8102
8x8B	0.9131
8x8C	0.9115
8x8D	0.9125
8x8E	0.9049
8x8F	0.9233
9x9A	0.9111
9x9B	0.9134
9x9C	0.9103
9x9D	0.9096
9x9E	0.9237
9x9F	0.9237
9x9G	0.9005
10x10A	0.9158
10x10B	0.9156
10x10C	0.9152
10x10D	0.9182
10x10E	0.8970

Table 6.4.8

MAXIMUM  $k_{eff}$  VALUES IN THE GENERIC BWR DAMAGED FUEL CONTAINER FOR A  
 MAXIMUM INITIAL ENRICHMENT OF 4.0 wt%  $^{235}\text{U}$  FOR DAMAGED FUEL AND 3.7  
 wt%  $^{235}\text{U}$  FOR INTACT FUEL

<b>Model Configuration inside the DFC</b>	<b>Maximum <math>k_{eff}</math></b>
Intact Assemblies (4 assemblies analyzed)	0.9241
Assemblies with missing rods (7 configurations analyzed)	0.9240
Assemblies with distributed enrichment (4 configurations analyzed)	0.9245
Collapsed Assemblies (6 configurations analyzed)	0.9258
Regular Arrays of Bare Fuel Rods (31 configurations analyzed)	0.9328

Table 6.4.9

MAXIMUM  $k_{\text{eff}}$  VALUES IN *THE MPC-24E/EF WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt%  $^{235}\text{U}$  AND NO SOLUBLE BORON.*

<b>Model Configuration inside the DFC</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
Intact Assemblies (2 assemblies analyzed)	0.9340
Assemblies with missing rods (4 configurations analyzed)	0.9350
Collapsed Assemblies (6 configurations analyzed)	0.9360
Regular Arrays of Bare Fuel Rods (36 configurations analyzed)	0.9480

Table 6.4.10

MAXIMUM  $k_{\text{eff}}$  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 (2600ppm)@ 5.0 %			
		Guide Tubes Filled, Moderator Density 0.93		Guide Tubes Voided, Moderator Density 1.00	
		1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>	1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>
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
16X16A	1900	0.9460	0.9450	0.9474	0.9434
17x17A	2600	0.9105	0.9145	0.9160	0.9161
17x17B	2600	0.9345	0.9358	0.9371	0.9356
17X17C	2600	0.9417	0.9431	0.9437	0.9430

Table 6.4.11

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 <sup>3</sup>	0.93 g/cm <sup>3</sup>	1.0 g/cm <sup>3</sup>	0.93 g/cm <sup>3</sup>
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
16X16A	1300	0.9468	0.9425	0.9433	0.9384
17x17A	1900	0.9105	0.9111	0.9106	0.9091
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

Table 6.4.12

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-24E/24EF WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt%  $^{235}\text{U}$  AND 600 PPM SOLUBLE BORON.

<i>Water Density inside the DFC</i>	<i>Bare Fuel Pellet Diameter</i>	<i>Maximum <math>k_{eff}</math></i>
1.00	minimum	0.9185
1.00	typical	0.9181
1.00	maximum	0.9171
0.95	typical	0.9145
0.90	typical	0.9125
0.60	typical	0.9063
0.10	typical	0.9025
0.02	typical	0.9025



Table 6.4.13

MAXIMUM  $k_{eff}$  VALUES IN THE MPC-32/32F WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt%  $^{235}\text{U}$ , 2900 PPM SOLUBLE BORON AND THE 15x15F ASSEMBLY CLASS AS INTACT ASSEMBLY.

<i>Water Density inside the DFC</i>	<i>Bare Fuel Pellet Diameter</i>	<i>Maximum <math>k_{eff}</math></i>
1.00	minimum	0.9374
1.00	typical	0.9372
1.00	maximum	0.9373
0.95	typical	0.9369
0.90	typical	0.9365
0.60	typical	0.9308
0.10	typical	0.9295
0.02	typical	0.9283

Table 6.4.14

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR THE MPC-32 AND MPC-32F WITH UP TO 8 DFCs UNDER VARIOUS MODERATION CONDITIONS.

<i>Fuel Assembly Class of Intact Fuel</i>	<i>Initial Enrichment (wt% <sup>235</sup>U)</i>	<i>Minimum Soluble Boron Content (ppm)</i>	<i>Maximum <math>k_{eff}</math></i>			
			<i>Filled Guide Tubes</i>		<i>Voided Guide Tubes</i>	
			<i>1.0 g/cm<sup>3</sup></i>	<i>0.93 g/cm<sup>3</sup></i>	<i>1.0 g/cm<sup>3</sup></i>	<i>0.93 g/cm<sup>3</sup></i>
<i>14x14A through 14x14E</i>	<i>4.1</i>	<i>1500</i>	<i>0.9277</i>	<i>0.9283</i>	<i>0.9336</i>	<i>0.9298</i>
	<i>5.0</i>	<i>2300</i>	<i>0.9139</i>	<i>0.9180</i>	<i>0.9269</i>	<i>0.9262</i>
<i>15x15A, B, C, G</i>	<i>4.1</i>	<i>1900</i>	<i>0.9345</i>	<i>0.9350</i>	<i>0.9350</i>	<i>0.9326</i>
	<i>5.0</i>	<i>2700</i>	<i>0.9307</i>	<i>0.9346</i>	<i>0.9347</i>	<i>0.9365</i>
<i>15x15D, E, F, H</i>	<i>4.1</i>	<i>2100</i>	<i>0.9322</i>	<i>0.9336</i>	<i>0.9340</i>	<i>0.9329</i>
	<i>5.0</i>	<i>2900</i>	<i>0.9342</i>	<i>0.9375</i>	<i>0.9385</i>	<i>0.9397</i>
<i>16x16A</i>	<i>4.1</i>	<i>1500</i>	<i>0.9322</i>	<i>0.9321</i>	<i>0.9335</i>	<i>0.9302</i>
	<i>5.0</i>	<i>2300</i>	<i>0.9198</i>	<i>0.9239</i>	<i>0.9289</i>	<i>0.9267</i>
<i>17x17A, B, C</i>	<i>4.1</i>	<i>2100</i>	<i>0.9284</i>	<i>0.9290</i>	<i>0.9294</i>	<i>0.9285</i>
	<i>5.0</i>	<i>2900</i>	<i>0.9308</i>	<i>0.9338</i>	<i>0.9355</i>	<i>0.9367</i>

Table 6.4.15

COMPARISON OF MAXIMUM  $k_{eff}$  VALUES FOR DIFFERENT FIXED NEUTRON ABSORBER MATERIALS

<i>Case</i>	<i>Maximum <math>k_{eff}</math></i>		<i>Reactivity Difference</i>
	<i>BORAL</i>	<i>METAMIC</i>	
<i>MPC-68, Intact Assemblies</i>	<i>0.9457</i>	<i>0.9452</i>	<i>-0.0005</i>
<i>MPC-68, with 16 DFCs</i>	<i>0.9328</i>	<i>0.9315</i>	<i>-0.0013</i>
<i>MPC-68F with 68 DFCs</i>	<i>0.8021</i>	<i>0.8019</i>	<i>-0.0002</i>
<i>MPC-24, 0ppm</i>	<i>0.9478</i>	<i>0.9491</i>	<i>+0.0013</i>
<i>MPC-24, 400ppm</i>	<i>0.9447</i>	<i>0.9457</i>	<i>+0.0010</i>
<i>MPC-24E, Intact Assemblies, 0ppm</i>	<i>0.9468</i>	<i>0.9494</i>	<i>+0.0026</i>
<i>MPC-24E, Intact Assemblies, 300ppm</i>	<i>0.9399</i>	<i>0.9410</i>	<i>+0.0011</i>
<i>MPC-24E, with 4 DFCs, 0ppm</i>	<i>0.9480</i>	<i>0.9471</i>	<i>-0.0009</i>
<i>MPC-32, Intact Assemblies, 1900ppm</i>	<i>0.9411</i>	<i>0.9397</i>	<i>-0.0014</i>
<i>MPC-32, Intact Assemblies, 2600ppm</i>	<i>0.9483</i>	<i>0.9471</i>	<i>-0.0012</i>
<i>Average Difference</i>			<i>+0.0001</i>

DELETED

FIGURE 6.4.1

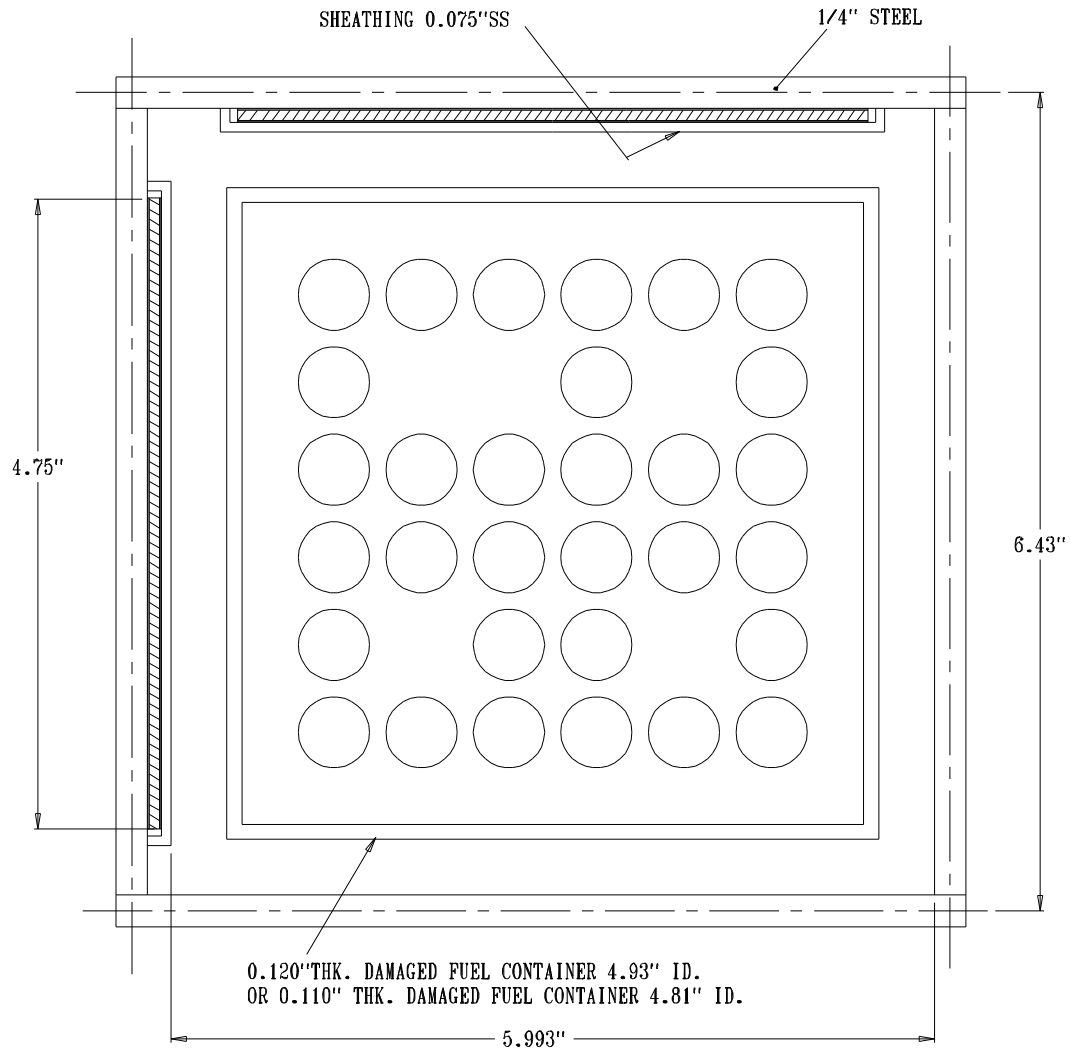


FIGURE 6.4.2; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH 6X6 ARRAY WITH 4 MISSING RODS IN THE MPC-68 BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

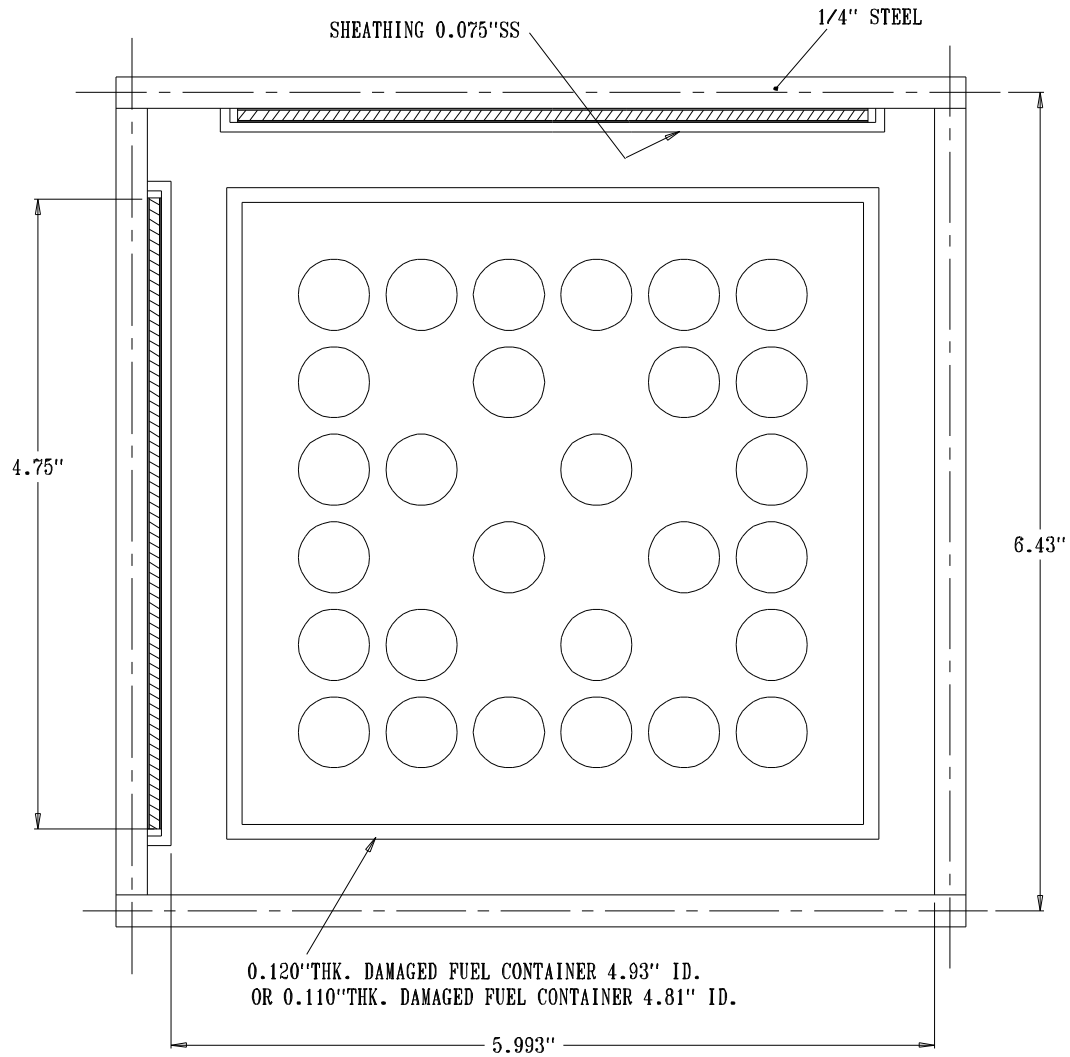
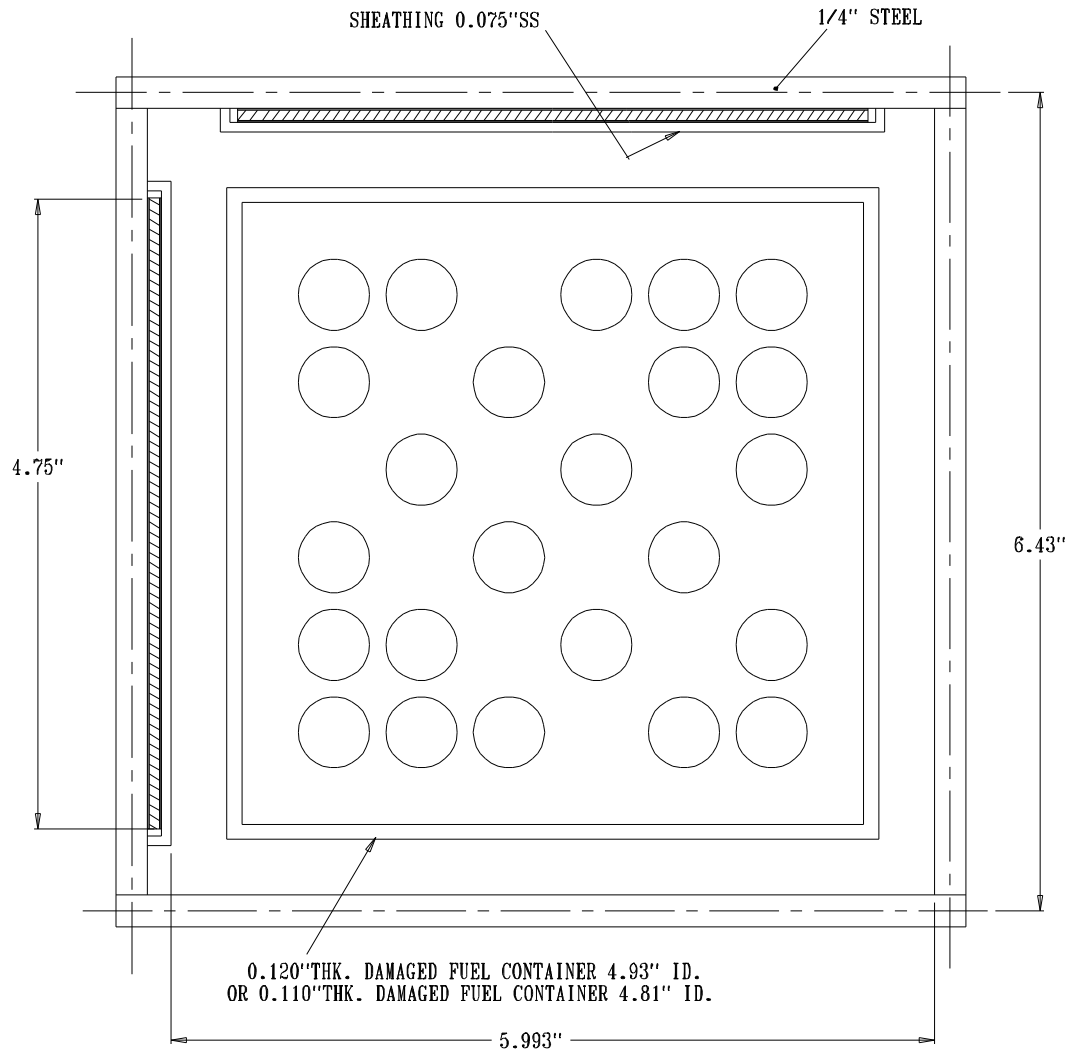


FIGURE 6.4.3; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.



0.120"THK. DAMAGED FUEL CONTAINER 4.93" ID.  
 OR 0.110"THK. DAMAGED FUEL CONTAINER 4.81" ID.

FIGURE 6.4.4; FAILED FUEL CALCULATION MODEL ( PLANAR CROSS-SECTION )  
 WITH 6X6 ARRAY WITH 12 MISSING RODS IN THE MPC-68 BASKET  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

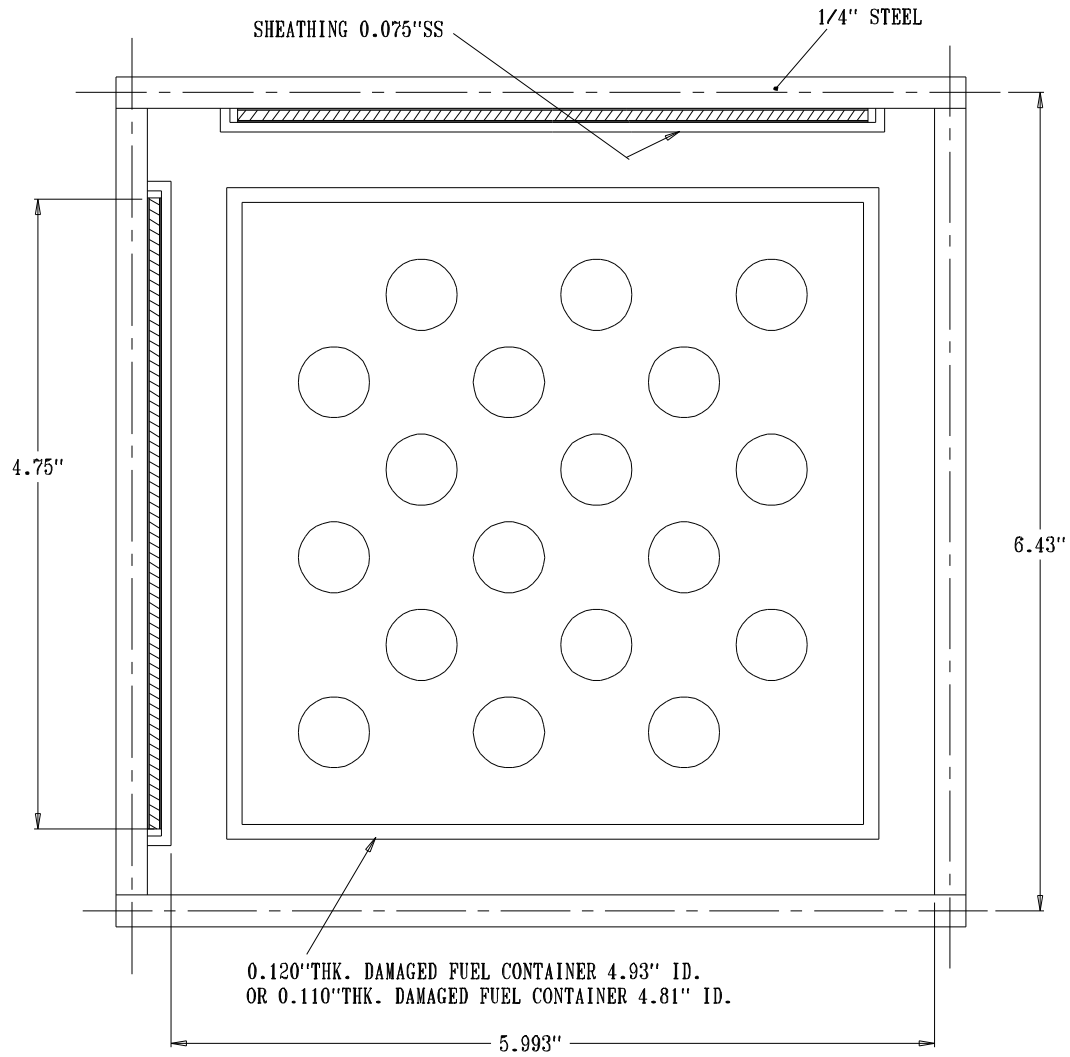


FIGURE 6.4.5; FAILED FUEL CALCULATION MODEL ( PLANAR CROSS-SECTION )  
WITH 6X6 ARRAY WITH 18 MISSING RODS IN THE MPC-68 BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.



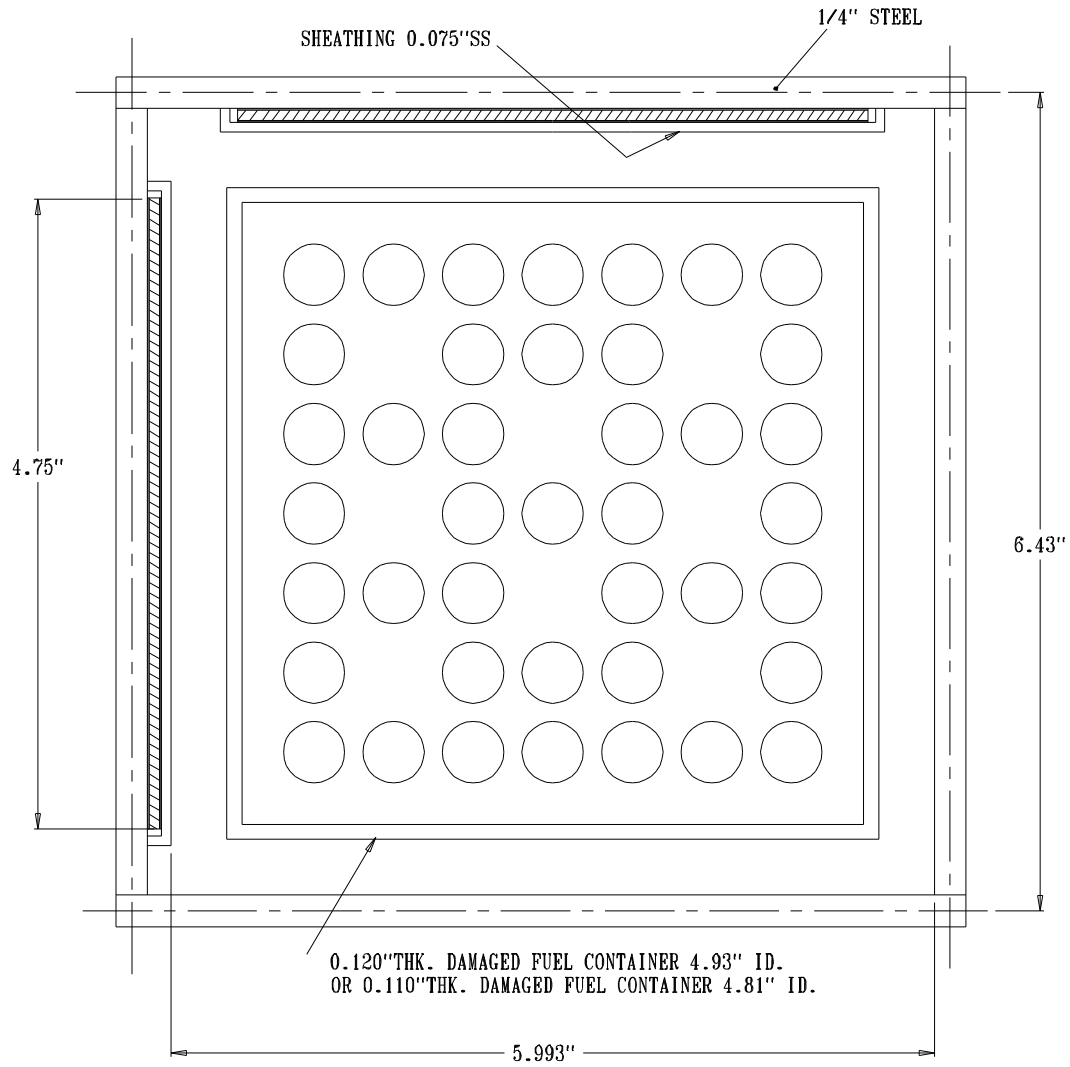


FIGURE 6.4.6; FAILED FUEL CALCULATION MODEL ( PLANAR CROSS-SECTION )  
WITH 7X7 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

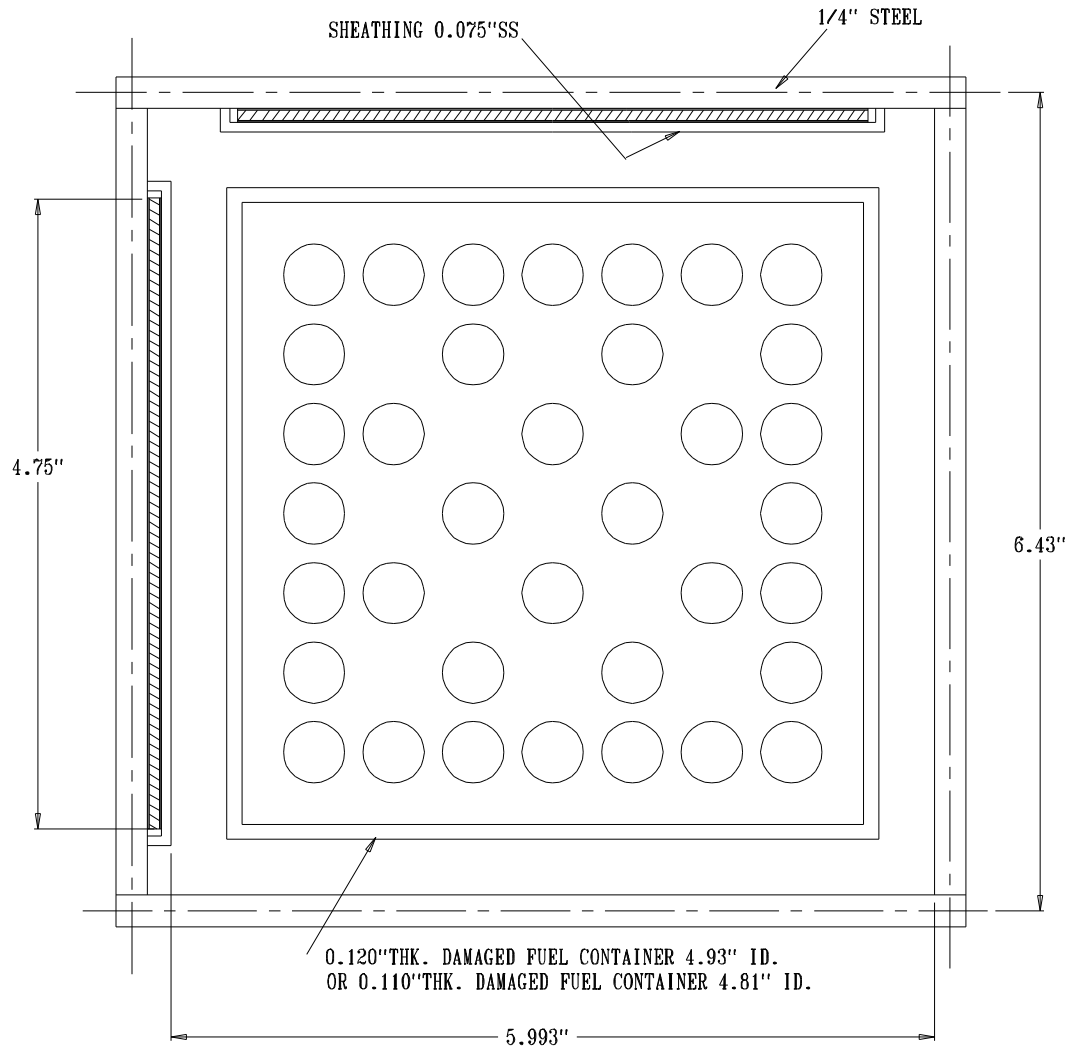


FIGURE 6.4.7; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH 7X7 ARRAY WITH 13 MISSING RODS IN THE MPC-68 BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

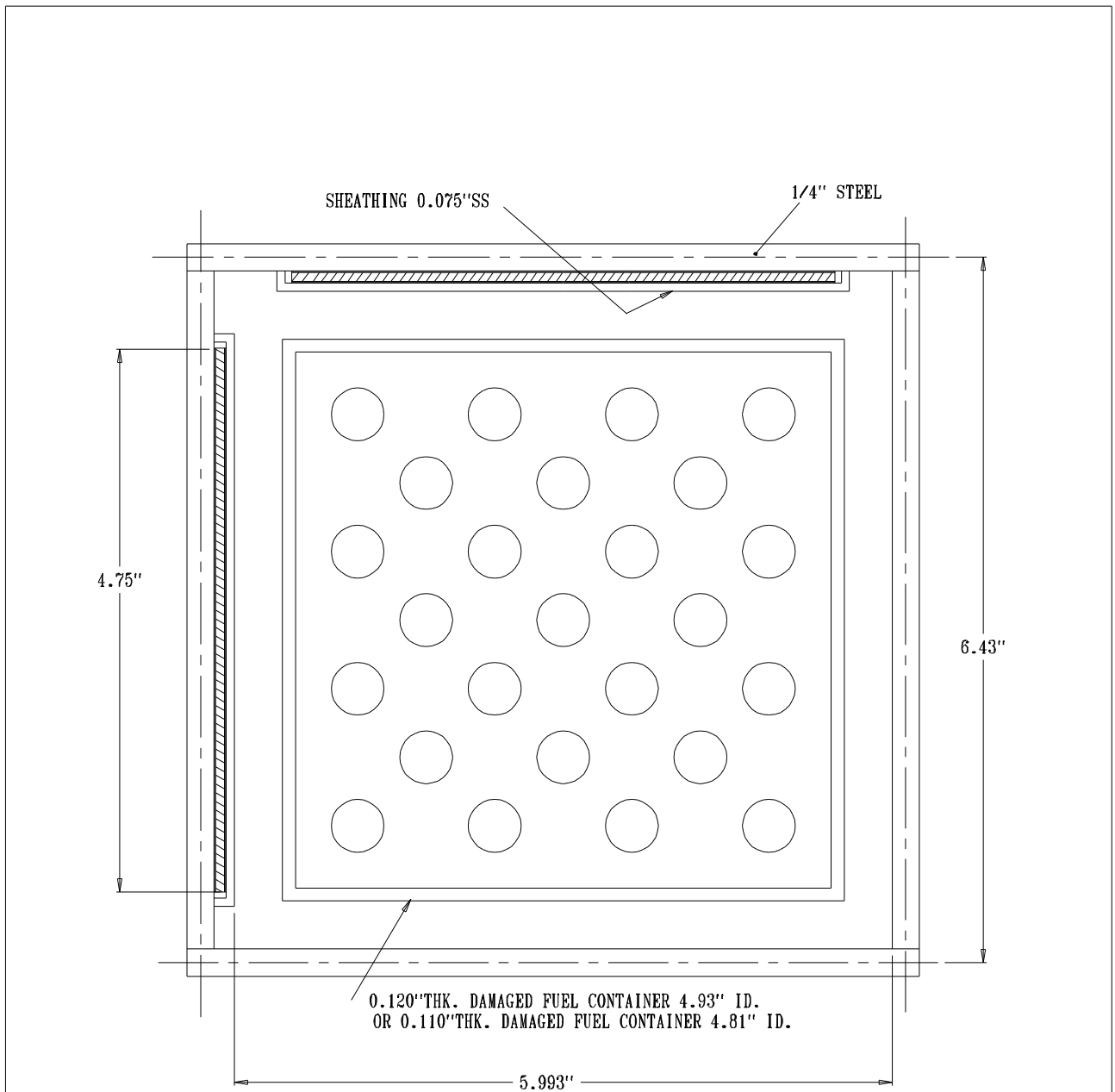
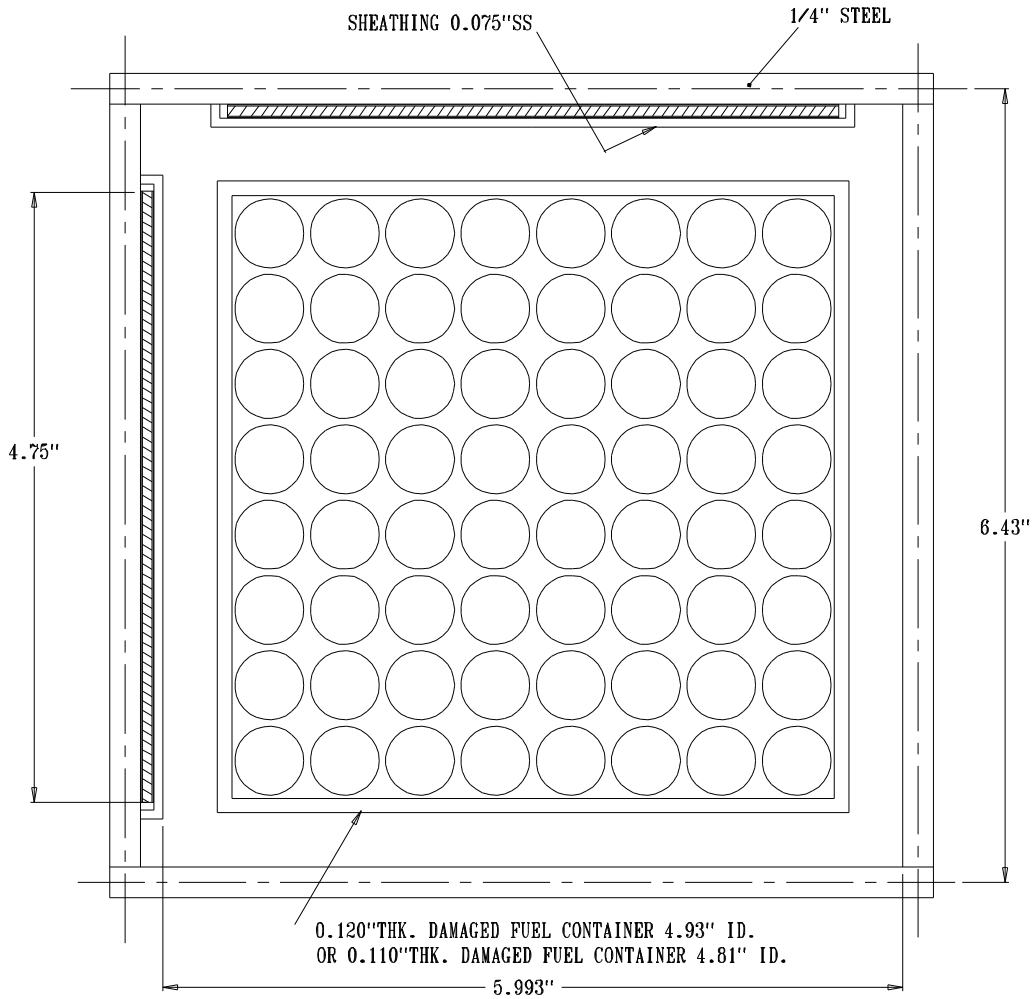


FIGURE 6.4.8; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION)  
WITH 7X7 ARRAY WITH 24 MISSING RODS IN THE MPC-68 BASKET  
( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )



**FIGURE 6.4.9; FAILED FUEL CALCULATION MODEL ( PLANAR CROSS-SECTION ) WITH DAMAGED FUEL COLLAPSED INTO 8X8 ARRAY IN THE MPC-68 BASKET ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )**

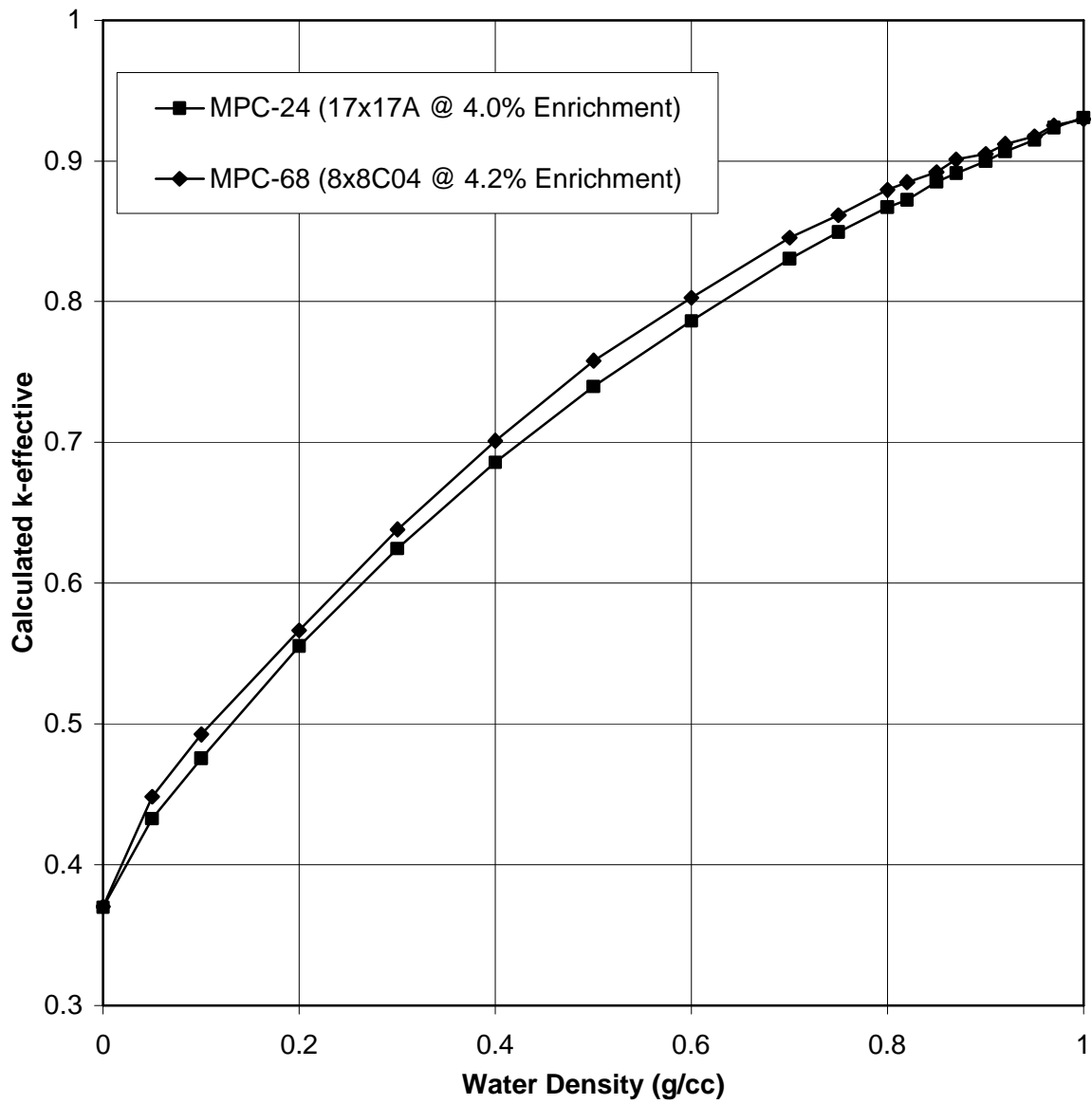


FIGURE 6.4.10; CALCULATED K-EFFECTIVE AS A FUNCTION OF INTERNAL MODERATOR DENSITY

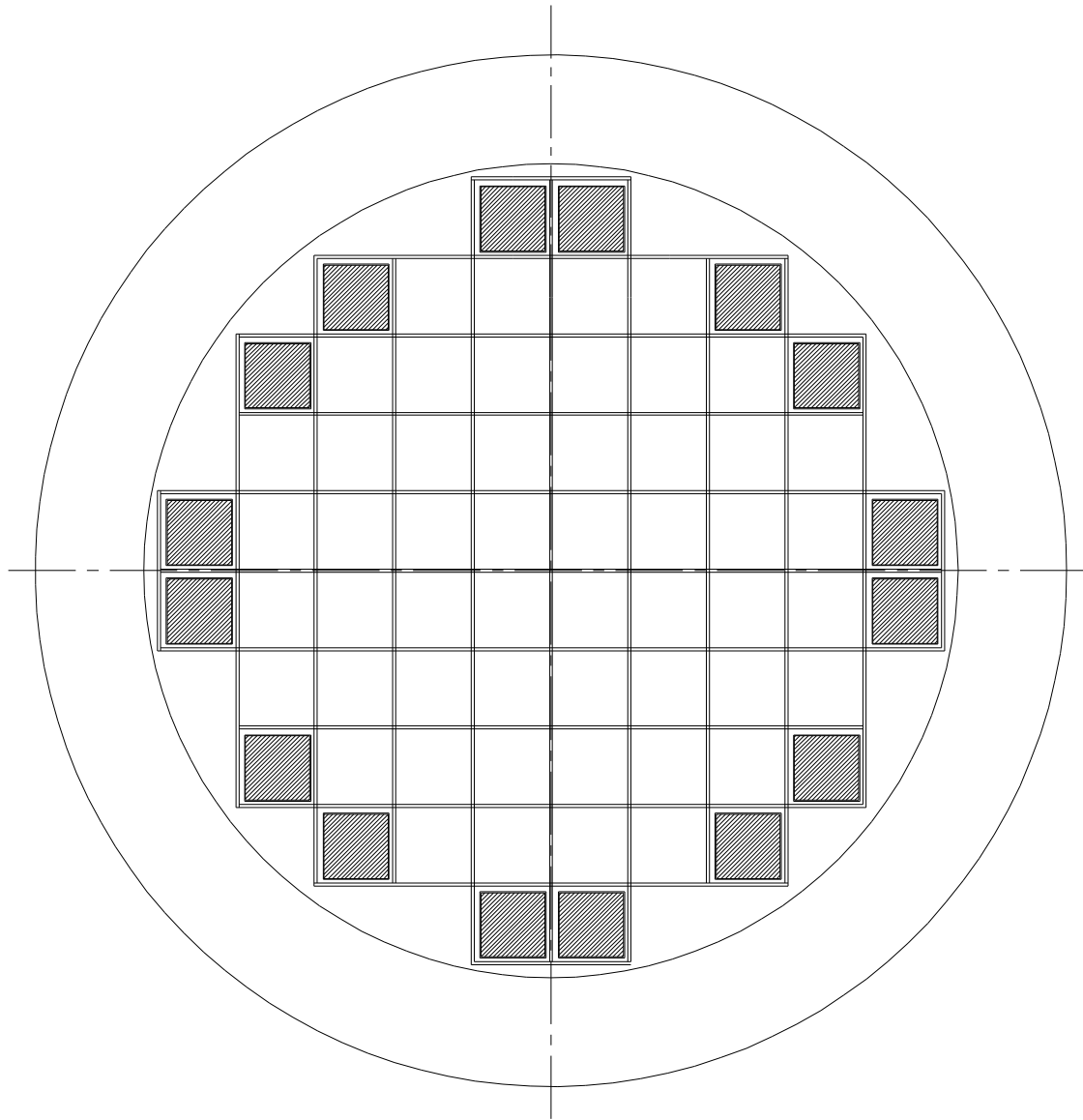


FIGURE 6.4.11; LOCATIONS OF THE DAMAGED FUEL CONTAINER  
IN THE MPC-68 AND MPC-68FF

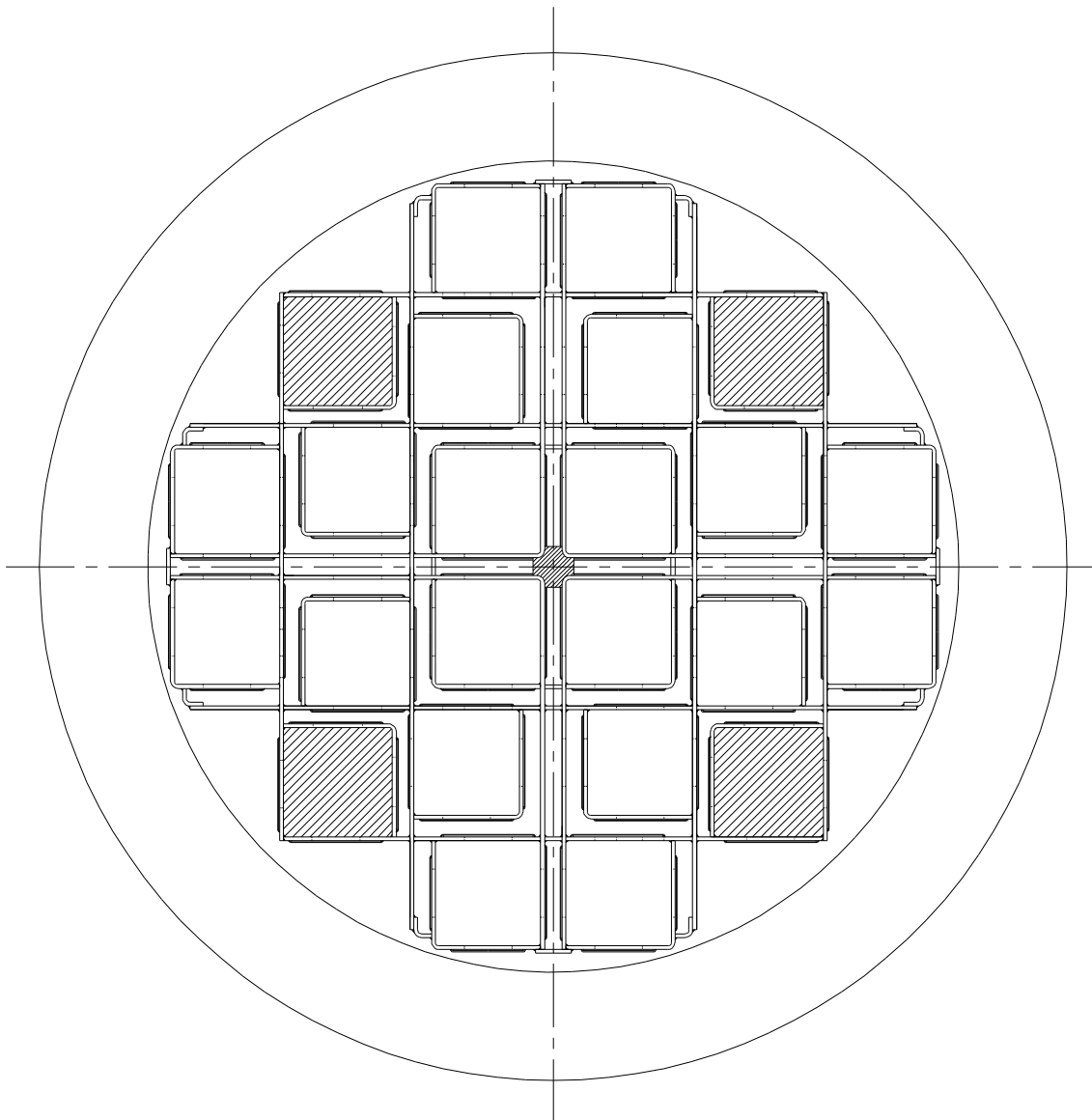


FIGURE 6.4.12; LOCATIONS OF THE DAMAGED FUEL CONTAINERS  
IN THE MPC 24E

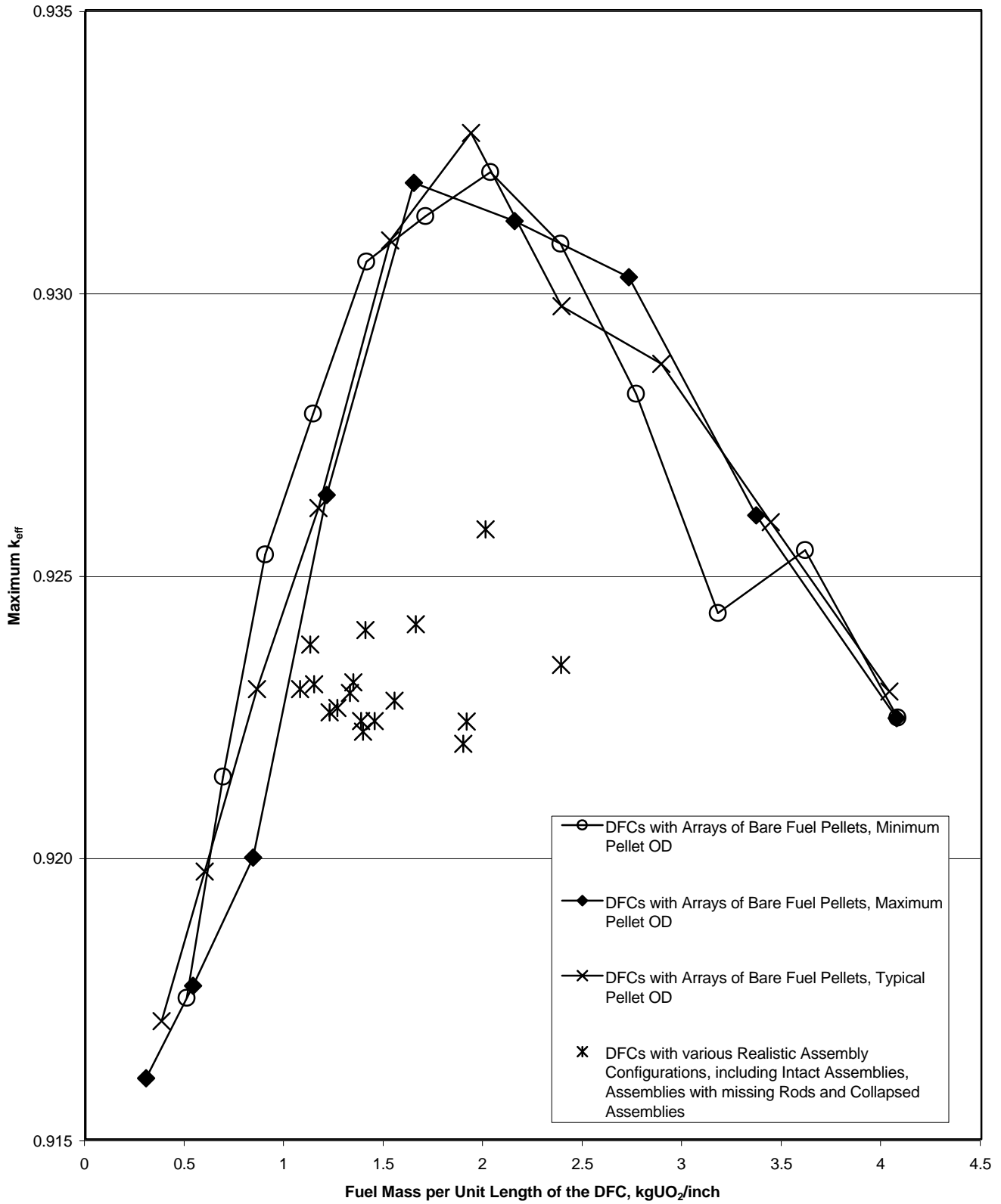


Figure 6.4.13: Maximum  $k_{eff}$  for the MPC-68 with Generic BWR Damaged Fuel Container, Initial Enrichment of 4.0 wt% for Damaged and 3.7 wt% for Intact Fuel.



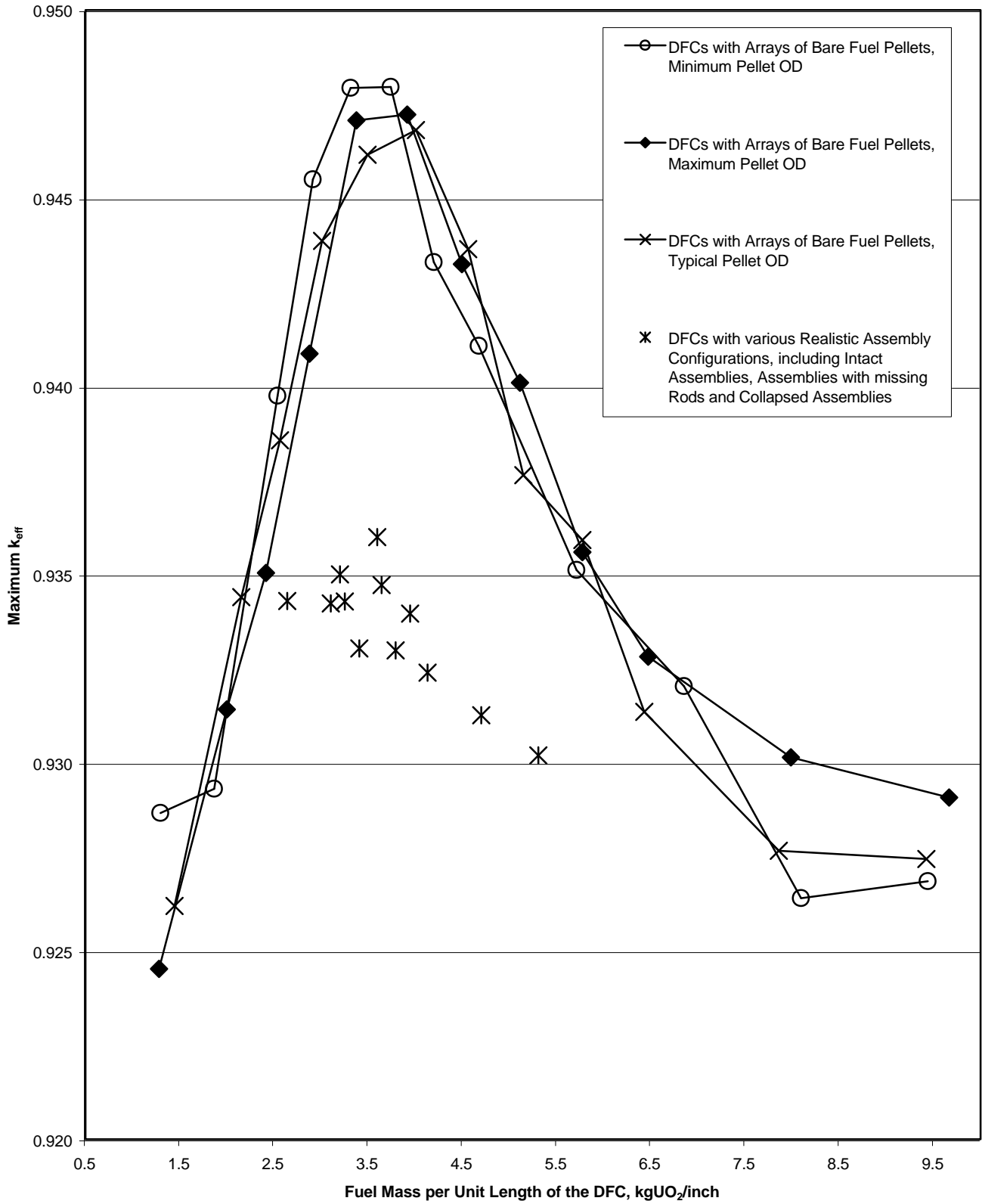


Figure 6.4.14: Maximum  $k_{eff}$  for the MPC-24E with Generic PWR Damaged Fuel Container, Initial Enrichment of 4.0 wt% for Damaged and Intact Fuel.

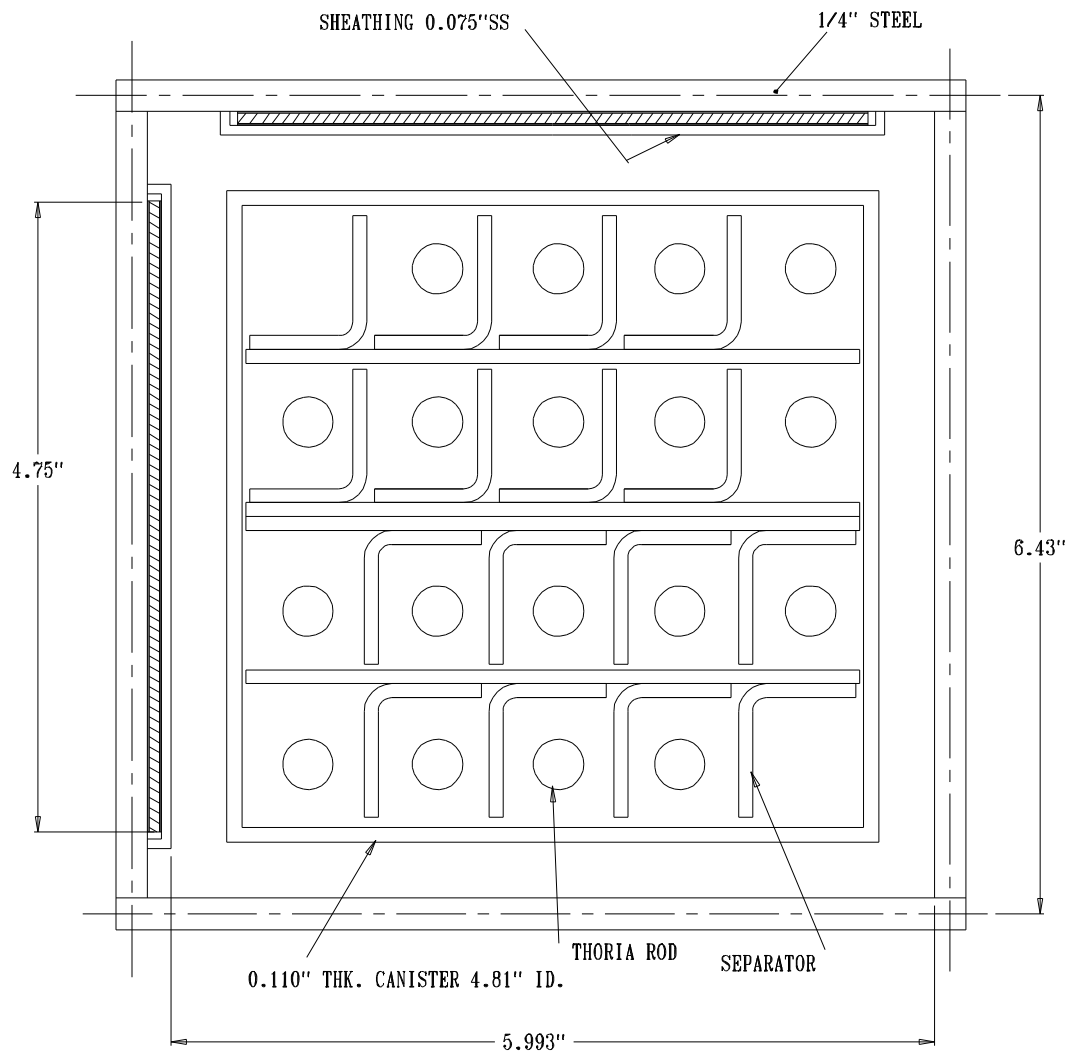


FIGURE 6.4.15; THORIA ROD CANISTER (PLANAR CROSS-SECTION)  
 WITH 18 THORIA RODS IN THE MPC-68 BASKET  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

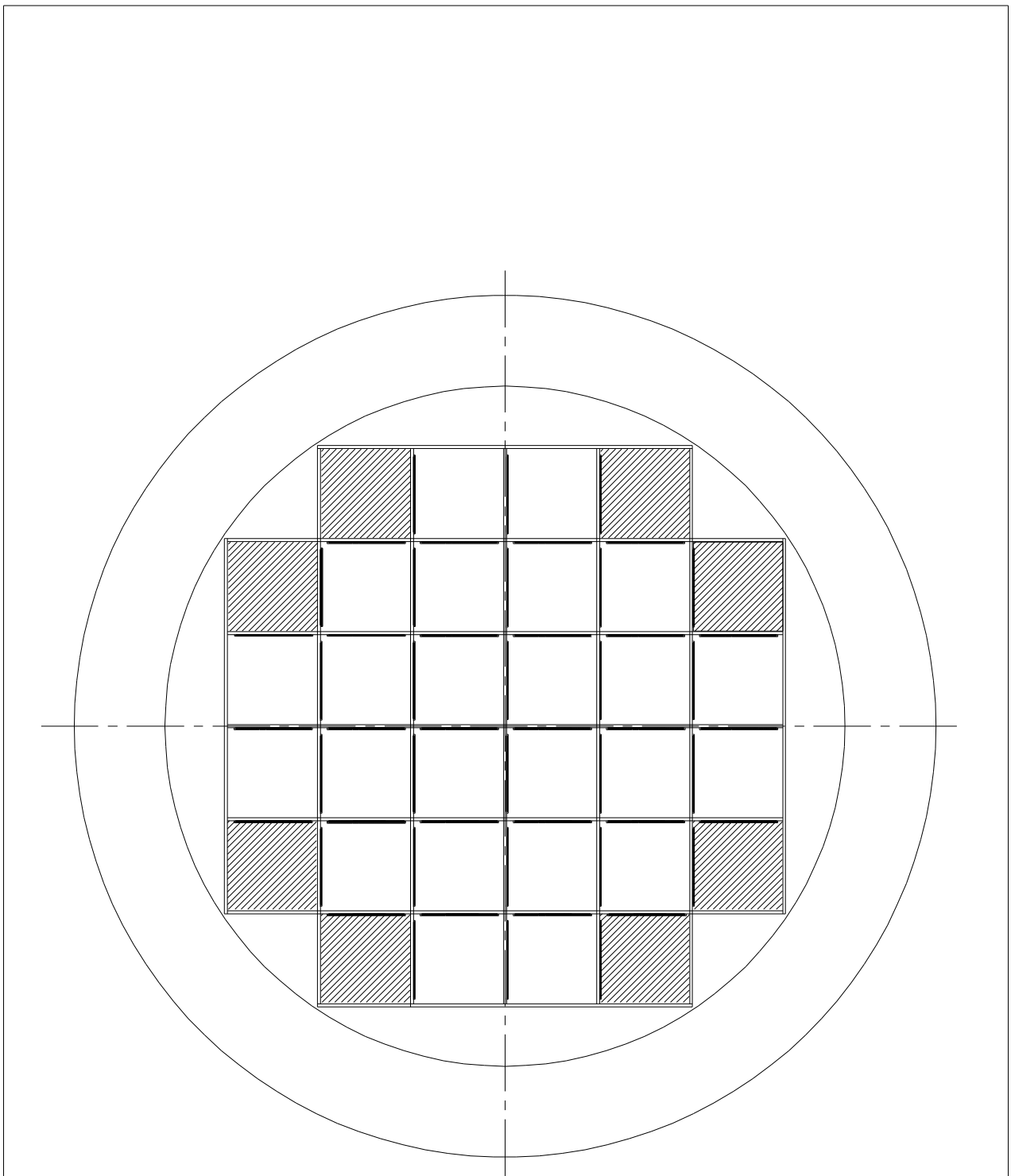


FIGURE 6.4.16; LOCATIONS OF THE DAMAGED FUEL CONTAINERS  
IN THE MPC-32.

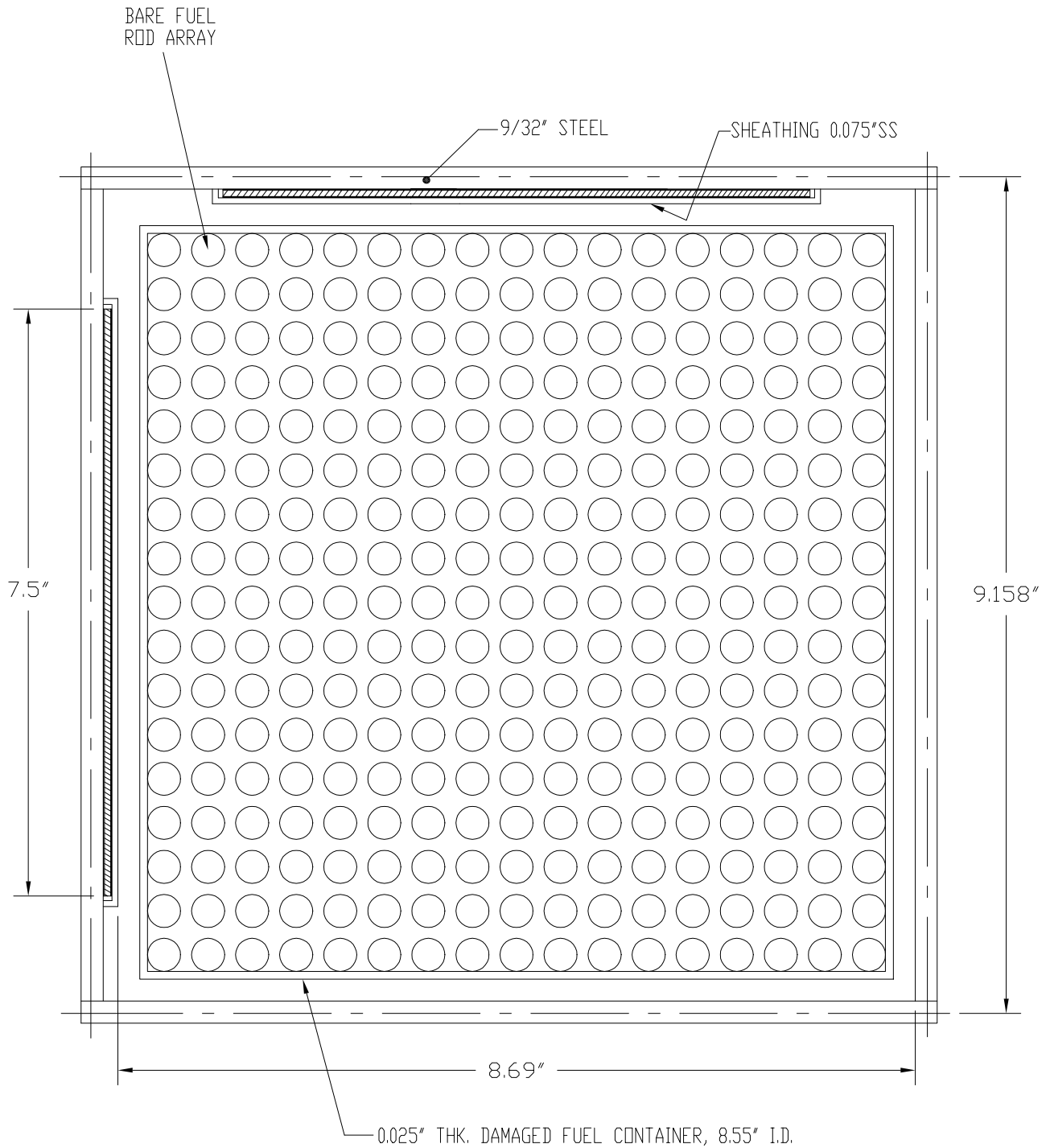


FIGURE 6.4.17; DAMAGED FUEL/FUEL DERIS CALCULATION MODEL (PLANAR CROSS-SECTION) WITH BARE FUEL RODS IN THE MPC-32 BASKET  
 ( SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS )

Benchmark calculations have been made on selected critical experiments, chosen, insofar as possible, to bound the range of variables in the cask designs. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24) or cell spacing (MPC-68), and (3) the  $^{10}\text{B}$  loading of the neutron absorber panels. Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included. No significant trends were evident in the benchmark calculations or the derived bias. Detailed benchmark calculations are presented in Appendix 6.A.

The benchmark calculations were performed with the same computer codes and cross-section data, described in Section 6.4, that were used to calculate the  $k_{\text{eff}}$  values for the cask. Further, all calculations were performed on the same computer hardware, specifically, personal computers using the pentium processor.

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STORM 100 System is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety, as well as the limiting fuel characteristics, are described in sufficient detail in this chapter to enable an evaluation of their effectiveness.

The HI-STORM 100 System is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons (Boral). An appraisal of the fixed neutron poisons has shown that they will remain effective for a storage period greater than 20 years, and there is no credible way to lose it, therefore there is no need to provide a positive means to verify their continued efficacy as required by 10CFR72.124(b).

The criticality evaluation has demonstrated that the cask will enable the storage of spent fuel for a minimum of 20 years with an adequate margin of safety. Further, the evaluation has demonstrated that the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, the HI-STORM 100 System is in full compliance with the double contingency requirements of 10CFR72.124. Therefore, it is concluded that the criticality design features for the HI-STORM 100 System are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The criticality evaluation provides reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel.

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## APPENDIX 6.A: BENCHMARK CALCULATIONS

### 6.A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the cask designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [6.A.1] is a continuous energy Monte Carlo code and KENO5a [6.A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [6.A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors<sup>†</sup> (trends) that have been reported (e.g., [6.A.3 through 6.A.5]) for calculations with collapsed cross section sets.

In cask designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the <sup>10</sup>B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included in the analyses.

Table 6.A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission". In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the energy of the average lethargy causing fission determined (post-processing).

Figures 6.A.1 and 6.A.2 show the calculated  $k_{\text{eff}}$  for the benchmark critical experiments as a function of the "energy of the average lethargy causing fission" for MCNP4a and KENO5a, respectively (UO<sub>2</sub> fuel only). The scatter in the data (even for comparatively minor variation in

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<sup>†</sup> Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

critical parameters) represents experimental error<sup>†</sup> in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 6.A.1 and 6.A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a  $k_{\text{eff}}$  of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a		
	Total	Truncated
MCNP4a	0.0009 ± 0.0011	0.0021 ± 0.0007
KENO5a	0.0030 ± 0.0012	0.0036 ± 0.0009

The values of bias shown in this table include both the bias derived directly from the calculated  $k_{\text{eff}}$  values in Table 6.A.1, and a more conservative value derived by arbitrarily truncating to 1.000 any calculated value that exceeds 1.000. The bias and standard error of the bias were calculated by the following equations<sup>††</sup>, with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [6.A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i=1}^n k_i \quad (6.A.1)$$

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<sup>†</sup> A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

<sup>††</sup> These equations may be found in any standard text on statistics, for example, reference [6.A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^n k_i^2 - (\sum_{i=1}^n k_i)^2 / n}{n(n-1)} \quad (6.A.2)$$

$$\text{Bias} = (1 - \bar{k}) \pm K\sigma_{\bar{k}} \quad (6.A.3)$$

where  $k_i$  are the calculated reactivities for  $n$  critical experiments;  $\sigma_{\bar{k}}$  is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); and  $K$  is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [6.A.18]).

Formula 6.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 6.A-2. The first portion of the equation,  $(1 - \bar{k})$ , is the actual bias which is added to the MCNP4a and KENO5a results. The second term,  $K\sigma_{\bar{k}}$ , which corresponds to  $\sigma_B$  in Section 6.4.3, is the uncertainty or standard error associated with the bias. The  $K$  values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual  $K$  values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The larger of the calculational biases (truncated bias) was used to evaluate the maximum  $k_{\text{eff}}$  values for the cask designs.

## 6.A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46% to 5.74% and therefore span the enrichment range for the MPC designs. Figures 6.A.3 and 6.A.4 show the calculated  $k_{\text{eff}}$  values (Table 6.A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, the MPC-68 configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 6.A.2 and Figure 6.A.5, confirm no significant difference in the calculated values of  $k_{\text{eff}}$  for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the

same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

### 6.A.3 Effect of $^{10}\text{B}$ Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the cask designs. Of these critical experiments, those performed by B&W are the most representative of the cask designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 6.A.3 lists the subset of experiments using thin neutron absorbers (from Table 6.A.1) and shows the reactivity worth ( $\Delta k$ ) of the absorber.<sup>†</sup>

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 6.A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with  $^{10}\text{B}$  concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 6.A.6 and Table 6.A.4 for the MPC-68 cask<sup>††</sup> geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

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<sup>†</sup> The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental ( $\Delta k$ ) change in reactivity due to the absorber.

<sup>††</sup> The MPC-68 geometry was chosen for this comparison since it contains the greater number of Boral panels and would therefore be expected to be the most sensitive to trends (errors) in calculations.

## 6.A.4 Miscellaneous and Minor Parameters

### 6.A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.<sup>†</sup> Analysis of these critical experiments are listed in Table 6.A.5 (subset of data in Table 6.A.1). There appears to be a small tendency toward overprediction of  $k_{\text{eff}}$  at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the cask calculations may be slightly more conservative than otherwise.

### 6.A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the cask designs, the fuel pellet diameters range from 0.303 to 0.3835 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.498 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of the fuel in the MPC designs. Based on the data in Table 6.A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments or the cask designs.

### 6.A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 6.A.6. Analyses of the very high boron concentration experiments (>1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the MPC-32 with various soluble boron concentration could be slightly conservative for the high soluble boron concentration.

## 6.A.5 MOX Fuel

The number of critical experiments with PuO<sub>2</sub> bearing fuel (MOX) is more limited than for UO<sub>2</sub> fuel. However, a number of MOX critical experiments have been analyzed and the results are

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<sup>†</sup>Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec cask design. A lead reflector is also not directly pertinent, but might be used in future designs.

shown in Table 6.A.7. Results of these analyses are generally above a  $k_{\text{eff}}$  of 1.00, indicating that when Pu is present, MCNP4a and KENO5a overpredict the reactivity.

This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggested that a small trend may exist with KENO5a. It is also possible that the overprediction in  $k_{\text{eff}}$  in both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated  $k_{\text{eff}}$  over a wide range of the spectral index (energy of the average lethargy causing fission).

#### 6.A.6 References

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**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

		<u>Calculated <math>k_{eff}</math></u>					<u>EALF (eV)</u>
	Reference	Identification	Enrich.	MCNP4a	KENO5a	MCNP4a	KENO5a
1	B&W-1484 (6.A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898 ± 0.0006	0.1759	0.1753
2	B&W-1484 (6.A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (6.A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (6.A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (6.A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (6.A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (6.A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (6.A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (6.A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (6.A.7)	Core XV <sup>††</sup>	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (6.A.7)	Core XVI <sup>††</sup>	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (6.A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	Calculated $k_{eff}$			EALF (eV)
				MCNP4a	KENO5a	MCNP4a	
13	B&W-1484 (6.A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708
14	B&W-1484 (6.A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (6.A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (6.A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (6.A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (6.A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (6.A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (6.A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (6.A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (6.A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (6.A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (6.A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	Calculated $k_{eff}$			EALF (eV)
				MCNP4a	KENO5a	MCNP4a	
25	French (6.A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018
27	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (6.A.11)	Steel Reflector, 3.912 cm separation	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (6.A.11)	Steel Reflector, 5.405 cm separation	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (6.A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	<u>Calculated <math>k_{eff}</math></u>				<u>EALF (eV)</u>
				MCNP4a	KENO5a	MCNP4a	KENO5a	
37	PNL-3626 (6.A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159	
38	PNL-3626 (6.A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044	
39	PNL-3626 (6.A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930	
40	PNL-3626 (6.A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854	
41	PNL-2615 (6.A.13)	Experiment 004/032 – no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159	
42	PNL-2615 (6.A.13)	Experiment 030 – Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154	
43	PNL-2615 (6.A.13)	Experiment 013 – Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164	
44	PNL-2615 (6.A.13)	Experiment 014 – Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164	
45	PNL-2615 (6.A.13)	Exp. 009 1.05% Boron Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162	
46	PNL-2615 (6.A.13)	Exp. 009 1.62% Boron Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173	
47	PNL-2615 (6.A.13)	Exp. 031 – Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171	
48	PNL-7167 (6.A.14)	Experiment 214R – with flux traps	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812	

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	Calculated $k_{eff}$			EALF (eV)
				MCNP4a	KENO5a	MCNP4a KENO5a	
49	PNL-7167 (6.A.14)	Experiment 214V3 –with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (6.A.15)	Case 173 – 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (6.A.15)	Case 177 – 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868
53	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1339	0.1165
56	WCAP-3385 (6.A.17)	Saxton Case 52 PuO <sub>2</sub> 0.52” pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (6.A.17)	Saxton Case 52 U 0.52” pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (6.A.17)	Saxton Case 56 PuO <sub>2</sub> 0.56” pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (6.A.17)	Saxton Case 56 borated PuO <sub>2</sub>	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (6.A.17)	Saxton Case 56 U 0.56” pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954

**Table 6.A.1**  
**Summary of Criticality Benchmark Calculations**

	Reference	Identification	Enrich.	Calculated $k_{eff}$			EALF (eV)	
				MCNP4a	KENO5a	MCNP4a	KENO5a	MCNP4a
61	WCAP-3385 (6.A.17)	Saxton Case 79 PuO <sub>2</sub> 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555	
62	WCAP-3385 (6.A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047	

Notes: NC stands for not calculated.

†

EALF is the energy of the average lethargy causing fission

††

The experimental results appear to be statistical outliers ( $>3\sigma$ ) suggesting the possibility of unusually large experimental error. Although they could be justifiably excluded, for conservatism, they were retained in determining the calculational basis.

Table 6.A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES<sup>†</sup>  
FOR VARIOUS ENRICHMENTS (UO<sub>2</sub>)

Enrichment	Calculated $k_{\text{eff}} \pm 1\sigma$	
	MCNP4a	KENO5a
3.0	$0.8465 \pm 0.0011$	$0.8478 \pm 0.0004$
3.5	$0.8820 \pm 0.0011$	$0.8841 \pm 0.0004$
3.75	$0.9019 \pm 0.0011$	$0.8987 \pm 0.0004$
4.0	$0.9132 \pm 0.0010$	$0.9140 \pm 0.0004$
4.2	$0.9276 \pm 0.0011$	$0.9237 \pm 0.0004$
4.5	$0.9400 \pm 0.0011$	$0.9388 \pm 0.0004$

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<sup>†</sup> Based on the MPC-68 with the GE 8x8R

Table 6.A.3

MCNP4a CALCULATED REACTIVITIES FOR  
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS (UO<sub>2</sub>)

Ref.	Experiment		$\Delta k$ Worth of Absorber	MCNP4a Calculated $k_{\text{eff}}$	EALF <sup>†</sup> (eV)
6.A.13	PNL-2615	Boral Sheet	0.0139	0.9994 ± 0.0012	0.1165
6.A.7	BAW-1484	Core XX	0.0165	1.0008 ± 0.0011	0.1724
6.A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996 ± 0.0012	0.1161
6.A.7	BAW-1484	Core XIX	0.0202	0.9961 ± 0.0012	0.2103
6.A.7	BAW-1484	Core XXI	0.0243	0.9994 ± 0.0010	0.1544
6.A.7	BAW-1484	Core XVII	0.0519	0.9962 ± 0.0012	0.2083
6.A.11	PNL-3602	Boral Sheet	0.0708	0.9941 ± 0.0011	0.3135
6.A.7	BAW-1484	Core XV	0.0786	0.9910 ± 0.0011	0.2092
6.A.7	BAW-1484	Core XVI	0.0845	0.9935 ± 0.0010	0.1757
6.A.7	BAW-1484	Core XIV	0.1575	0.9953 ± 0.0011	0.2022
6.A.7	BAW-1484	Core XIII	0.1738	1.0020 ± 0.0011	0.1988
6.A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991 ± 0.0011	0.3722

<sup>†</sup> EALF is the energy of the average lethargy causing fission



Table 6.A.4  
 COMPARISON OF MCNP4a AND KENO5a  
 CALCULATED REACTIVITIES<sup>†</sup> FOR VARIOUS BORON LOADINGS (UO<sub>2</sub>)

<sup>10</sup> B, g/cm <sup>2</sup>	Calculated $k_{\text{eff}} \pm 1\sigma$	
	MCNP4a	KENO5a
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

<sup>†</sup> based on 4.5% enrichment GE 8x8R in the MPC-68 cask.

Table 6.A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH  
THICK LEAD AND STEEL REFLECTORS<sup>†</sup> (UO<sub>2</sub>)

Ref.	Case	Enrichment, wt%	Separation, cm	MCNP4a $k_{eff}$	KENO5a $k_{eff}$
6.A.11	Steel Reflector	2.35	1.321	$0.9980 \pm 0.0009$	$0.9992 \pm 0.0006$
		2.35	2.616	$0.9968 \pm 0.0009$	$0.9964 \pm 0.0006$
		2.35	3.912	$0.9974 \pm 0.0010$	$0.9980 \pm 0.0006$
		2.35	$\infty$	$0.9962 \pm 0.0008$	$0.9939 \pm 0.0006$
6.A.11	Steel Reflector	4.306	1.321	$0.9997 \pm 0.0010$	$1.0012 \pm 0.0007$
		4.306	2.616	$0.9994 \pm 0.0012$	$0.9974 \pm 0.0007$
		4.306	3.405	$0.9969 \pm 0.0011$	$0.9951 \pm 0.0007$
		4.306	$\infty$	$0.9910 \pm 0.0020$	$0.9947 \pm 0.0007$
6.A.11	Lead Reflector	4.306	0.55	$1.0025 \pm 0.0011$	$0.9997 \pm 0.0007$
		4.306	1.956	$1.0000 \pm 0.0012$	$0.9985 \pm 0.0007$
		4.306	5.405	$0.9971 \pm 0.0012$	$0.9946 \pm 0.0007$

<sup>†</sup> Arranged in order of increasing reflector fuel spacing.

Table 6.A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE  
BORON CONCENTRATIONS (UO<sub>2</sub>)

Reference	Experiment	Boron Concentration ppm	Calculated $k_{\text{eff}}$	
			MCNP4a	KENO5a
6.A.15	PNL-4267	0	$0.9974 \pm 0.0012$	-
6.A.8	BAW-1645-4	886	$0.9970 \pm 0.0010$	$0.9924 \pm 0.0006$
6.A.9	BAW-1810	1337	$1.0023 \pm 0.0010$	-
6.A.9	BAW-1810	1899	$1.0060 \pm 0.0009$	-
6.A.15	PNL-4267	2550	$1.0057 \pm 0.0010$	-

Table 6.A.7

## CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case <sup>†</sup>	MCNP4a		KENO 5a	
		k <sub>eff</sub>	EALF <sup>††</sup> (eV)	k <sub>eff</sub>	EALF <sup>††</sup> (eV)
PNL-5803 [6.A.16]	MOX Fuel – Exp No 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel – Exp No 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel – Exp No 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel – Exp No 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP- 3385- 54 [6.A.17]	Saxton @ 0.52” pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56” pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56” pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79” pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

<sup>†</sup> Arranged in order of increasing lattice spacing.

<sup>††</sup> EALF is the energy of the average lethargy causing fission.

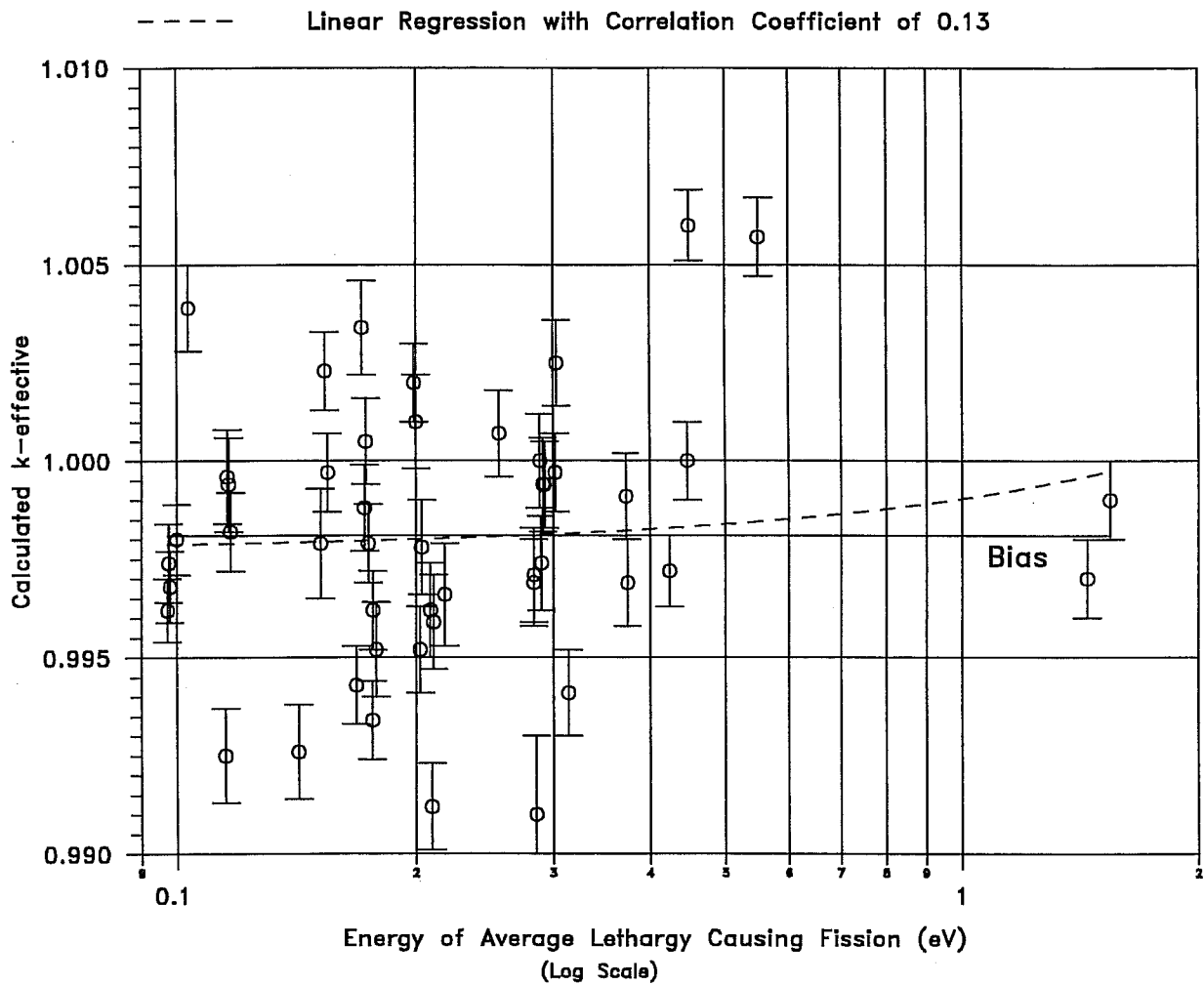


FIGURE 6.A.1 MCNP4a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

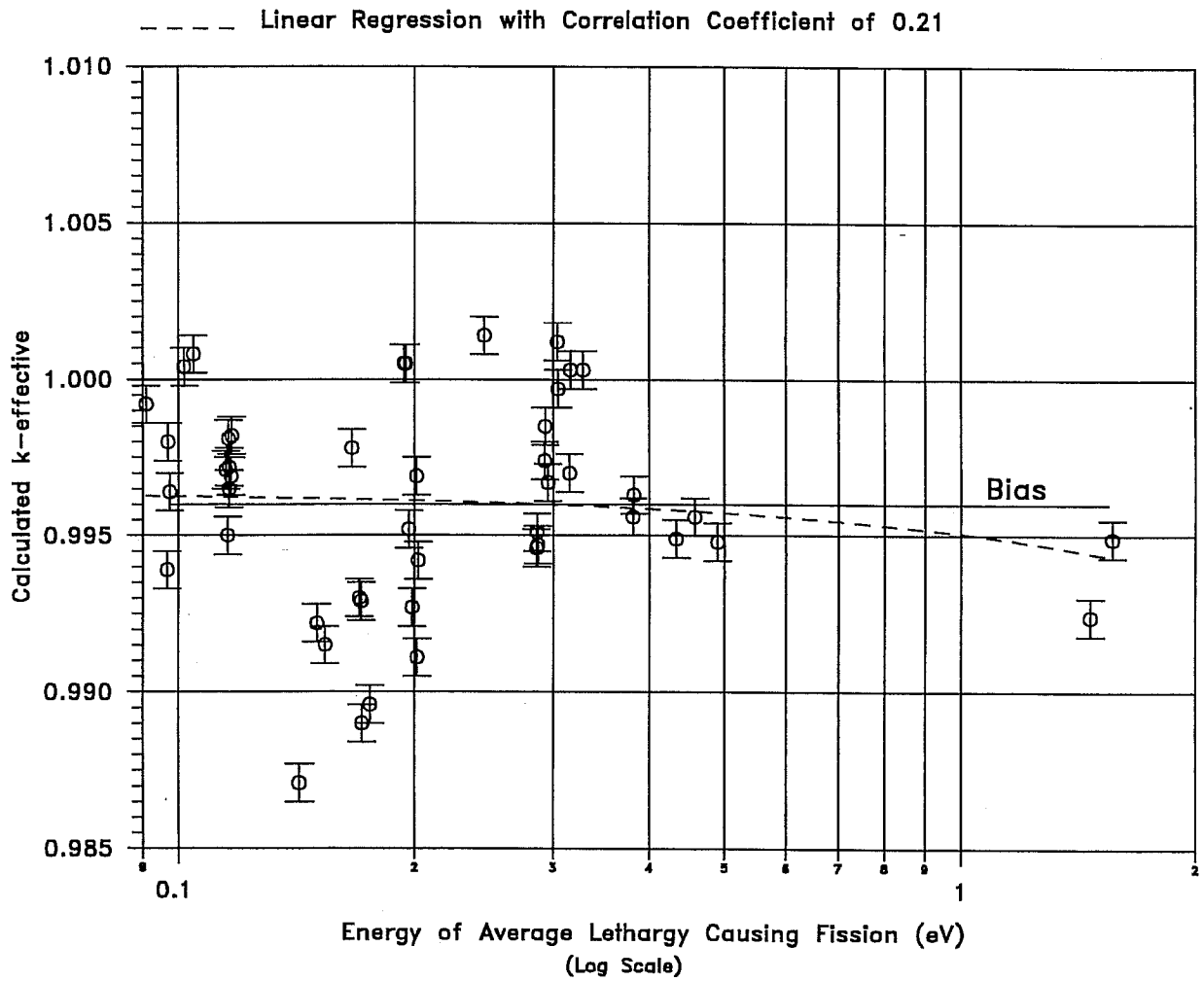


FIGURE 6.A.2 KEN05a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

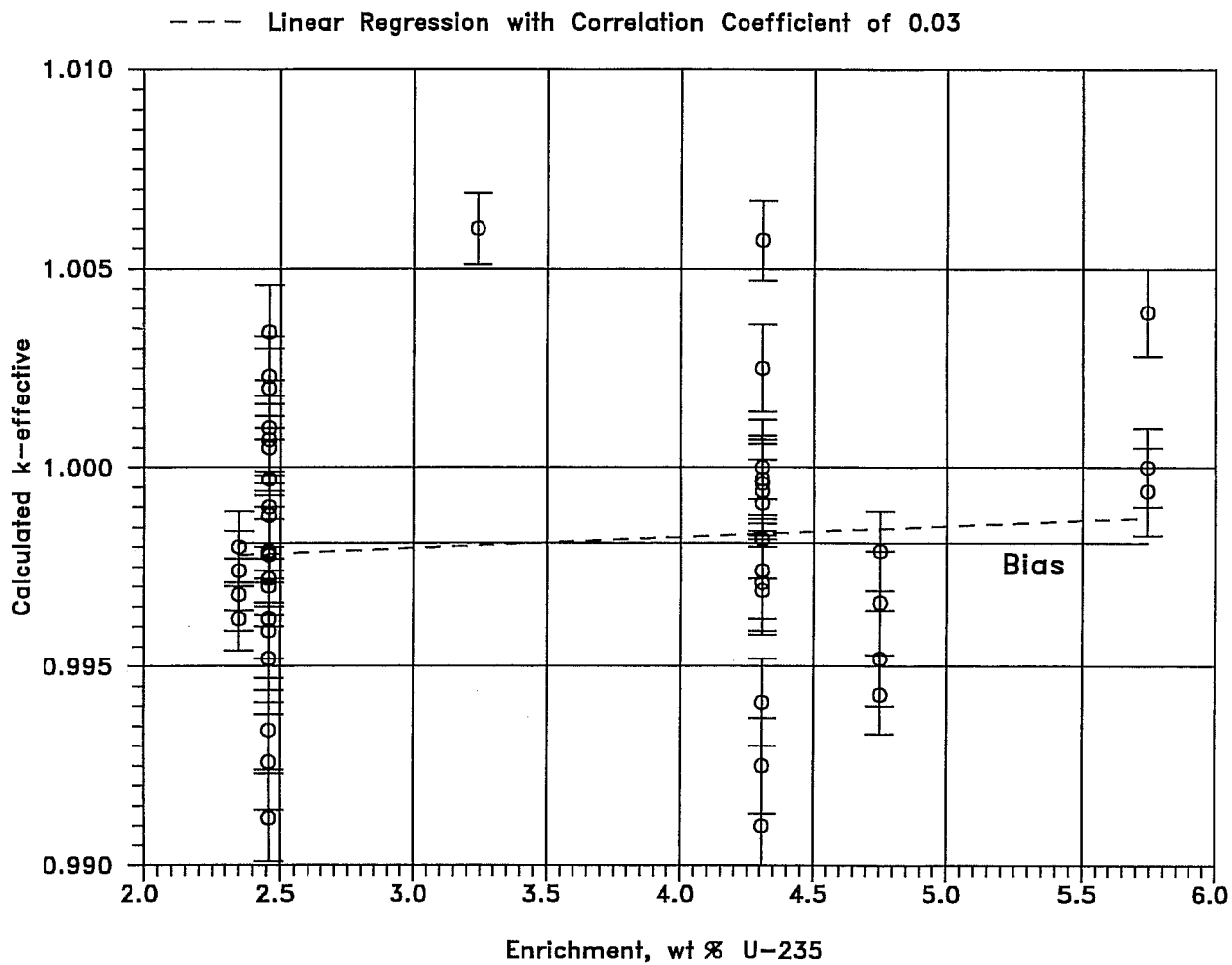


FIGURE 6.A.3 MCNP4a CALCULATED  $k$ -eff VALUES AT VARIOUS U-235 ENRICHMENTS

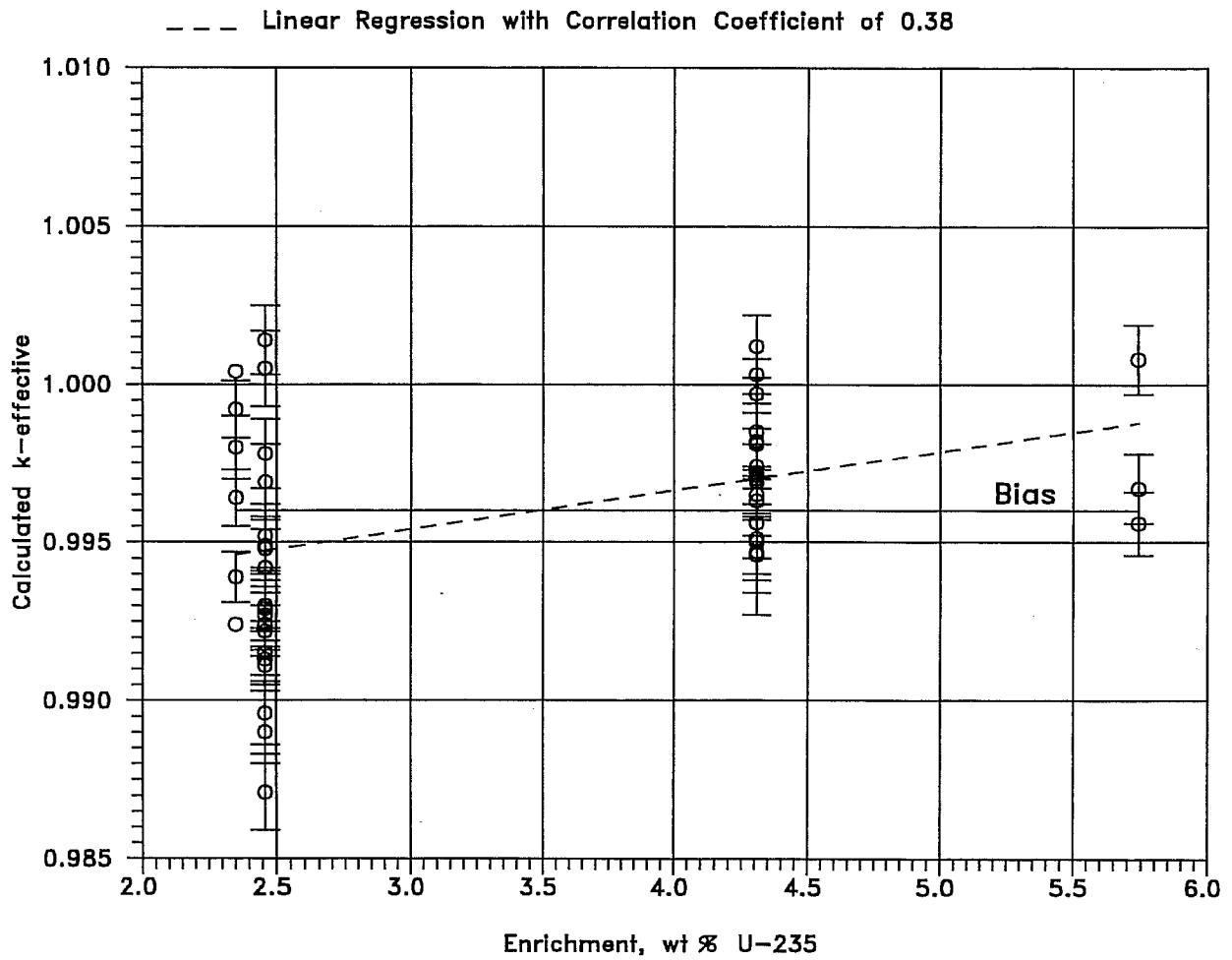


FIGURE 6.A.4 KENO5a CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS



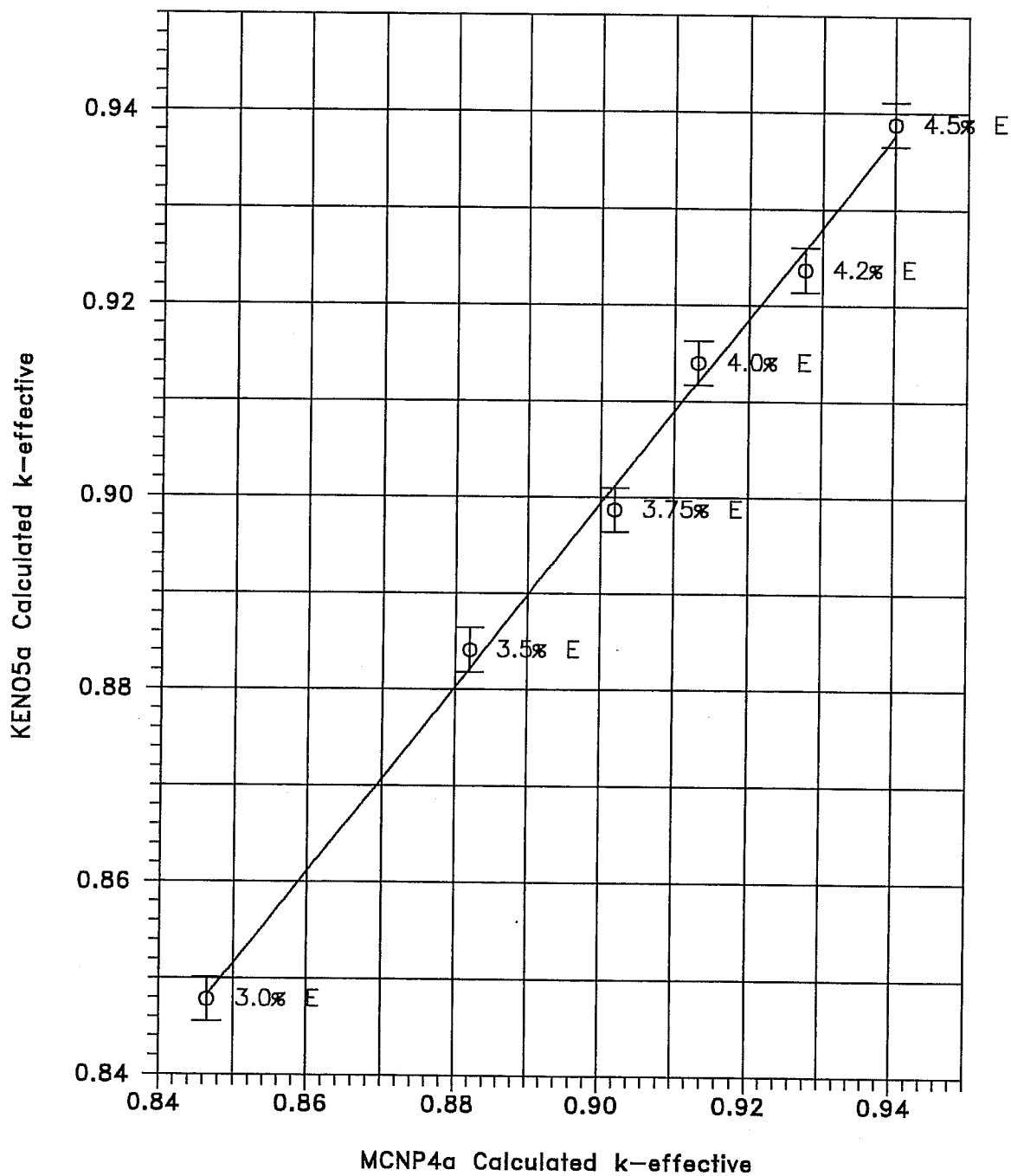


FIGURE 6.A.5 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

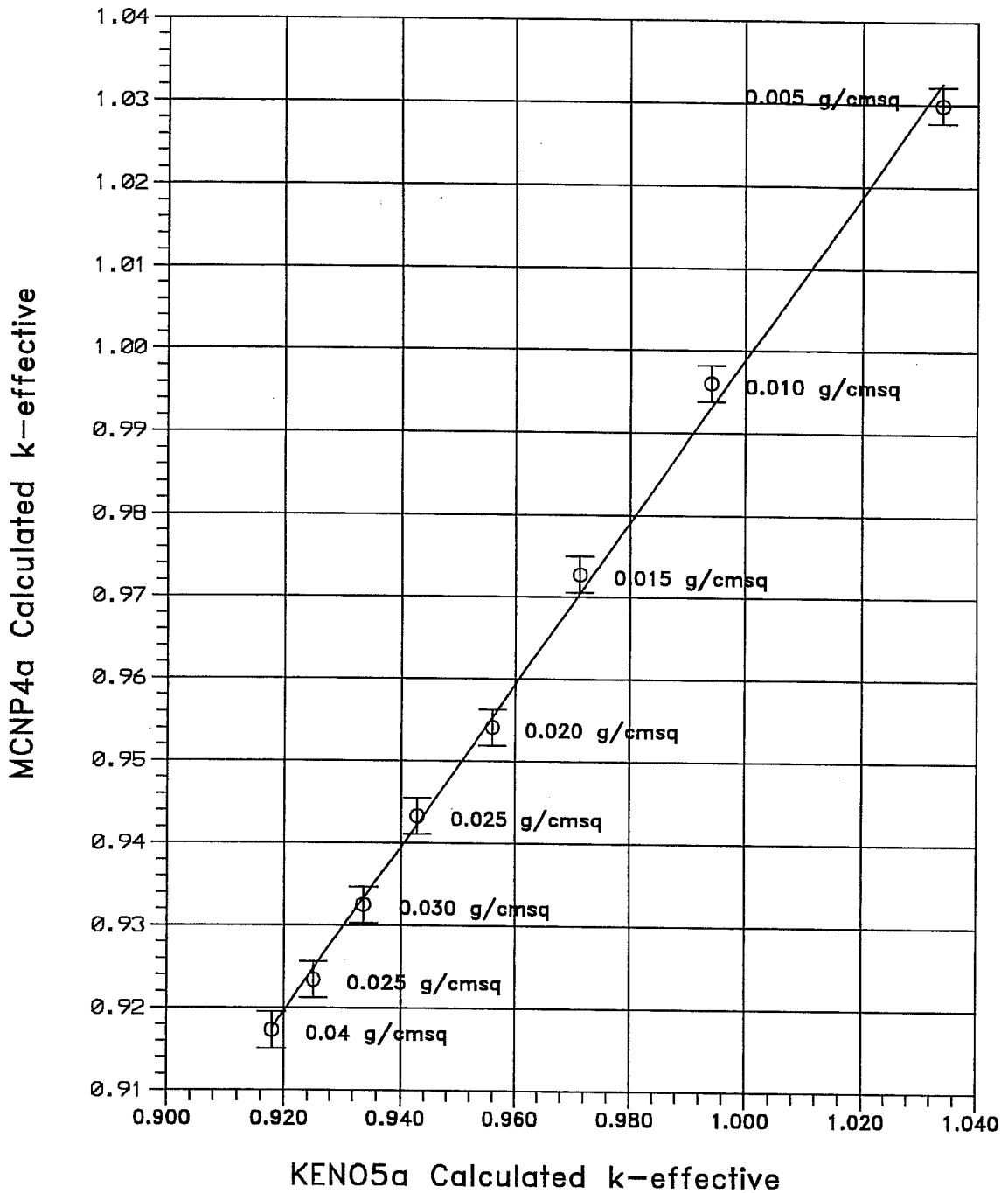


FIGURE 6.A.6 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

## APPENDIX 6.B: DISTRIBUTED ENRICHMENTS IN BWR FUEL

Fuel assemblies used in BWRs utilize fuel rods of varying enrichments as a means of controlling power peaking during in-core operation. For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations have been performed to confirm that this statement remains valid in the geometry of the MPC-68. These calculations are based on fuel assembly designs currently in use and two hypothetical distributions, all intended to illustrate that calculations with uniform average enrichments are conservative.

The average enrichment is calculated as the linear average of the various fuel rod enrichments, i.e.,

$$\bar{E} = \frac{1}{n} \sum_{i=1}^n E_i,$$

where  $E_i$  is the enrichment in each of the  $n$  rods, and  $\bar{E}$  is the assembly average enrichment. This parameter conservatively characterizes the fuel assembly and is readily available for specific fuel assemblies in determining the acceptability of the assembly for placement in the MPC-68 cask.

The criticality calculations for average and distributed enrichment cases are compared in Table 6.B.1 to illustrate and confirm the conservatism inherent in using average enrichments. With two exceptions, the cases analyzed represent realistic designs currently in use and encompass fuel with different ratios of maximum pin enrichment to average assembly enrichment. The two exceptions are hypothetical cases intended to extend the models to higher enrichments and to demonstrate that using the average enrichment remains conservative.

Table 6.B.1 shows that, in all cases, the averaged enrichment yields conservative values of reactivity relative to distributed enrichments for both the actual fuel designs and the hypothetical higher enrichment cases. Thus, it is concluded that uniform average enrichments will always yield higher (more conservative) values for reactivity than the corresponding distributed enrichments.<sup>†</sup>

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<sup>†</sup> This conclusion implicitly assumes the higher enrichment fuel rods are located internal to the assembly (as in BWR fuel), and the lower enriched rods are on the outside.

Table 6.B.1

**COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND  
DISTRIBUTED ENRICHMENTS**

<b>Case</b>	<b>Average %E</b>	<b>Peak Rod E%</b>	<b>Calculated <math>k_{eff}</math></b>	
			<b>Average E</b>	<b>Distributed E</b>
8x8C04	3.01	3.80	0.8549	0.8429
8x8C04	3.934	4.9	0.9128	0.9029
8x8D05	3.42	3.95	0.8790	0.8708
8x8D05	3.78	4.40	0.9030	0.8974
8x8D05	3.90	4.90	0.9062	0.9042
9x9B01	4.34	4.71	0.9347	0.9285
9x9D01	3.35	4.34	0.8793	0.8583
Hypothetical #1 (48 outer rods of 3.967%E, 14 inner rods of 5.0%)	4.20	5.00	0.9289	0.9151
Hypothetical #2 (48 outer rods of 4.354%E, 14 inner rods of 5.0%)	4.50	5.00	0.9422	0.9384

## APPENDIX 6.C: CALCULATIONAL SUMMARY

The following table lists the maximum  $k_{\text{eff}}$  (including bias, uncertainties, and calculational statistics), MCNP calculated  $k_{\text{eff}}$ , standard deviation, and energy of average lethargy causing fission (EALF) for each of the candidate fuel types and basket configurations.

Table 6.C.1  
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
AND BASKET CONFIGURATIONS

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{\text{eff}}</math></b>	<b>Calculated <math>k_{\text{eff}}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A01	HI-STAR	0.9295	0.9252	0.0008	0.2084
14x14A02	HI-STAR	0.9286	0.9242	0.0008	0.2096
14x14A03	HI-STORM	0.3080	0.3047	0.0003	3.37E+04
14x14A03	HI-TRAC	0.9283	0.9239	0.0008	0.2096
14x14A03	HI-STAR	0.9296	0.9253	0.0008	0.2093
14x14B01	HI-STAR	0.9159	0.9117	0.0007	0.2727
14x14B02	HI-STAR	0.9169	0.9126	0.0008	0.2345
14x14B03	HI-STAR	0.9110	0.9065	0.0009	0.2545
14x14B04	HI-STAR	0.9084	0.9039	0.0009	0.2563
B14x14B01	HI-TRAC	0.9237	0.9193	0.0008	0.2669
B14x14B01	HI-STAR	0.9228	0.9185	0.0008	0.2675
14x14C01	HI-TRAC	0.9273	0.9230	0.0008	0.2758
14x14C01	HI-STAR	0.9258	0.9215	0.0008	0.2729
14x14C02	HI-STAR	0.9265	0.9222	0.0008	0.2765
14x14C03	HI-TRAC	0.9274	0.9231	0.0008	0.2839
14x14C03	HI-STAR	0.9287	0.9242	0.0009	0.2825

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14D01	HI-TRAC	0.8531	0.8488	0.0008	0.3316
14x14D01	HI-STAR	0.8507	0.8464	0.0008	0.3308
14x14E01	HI-STAR	0.7598	0.7555	0.0008	0.3890
14x14E02	HI-TRAC	0.7627	0.7586	0.0007	0.3591
14x14E02	HI-STAR	0.7627	0.7586	0.0007	0.3607
14x14E03	HI-STAR	0.6952	0.6909	0.0008	0.2905
15x15A01	HI-TRAC	0.9205	0.9162	0.0008	0.2595
15x15A01	HI-STAR	0.9204	0.9159	0.0009	0.2608
15x15B01	HI-STAR	0.9369	0.9326	0.0008	0.2632
15C15B02	HI-STAR	0.9338	0.9295	0.0008	0.2640
15x15B03	HI-STAR	0.9362	0.9318	0.0008	0.2632
15x15B04	HI-STAR	0.9370	0.9327	0.0008	0.2612
15x15B05	HI-STAR	0.9356	0.9313	0.0008	0.2606
15x15B06	HI-STAR	0.9366	0.9324	0.0007	0.2638
B15x15B01	HI-TRAC	0.9387	0.9344	0.0008	0.2616
B15x15B01	HI-STAR	0.9388	0.9343	0.0009	0.2626
15x15C01	HI-STAR	0.9255	0.9213	0.0007	0.2493
15x15C02	HI-STAR	0.9297	0.9255	0.0007	0.2457
15x15C03	HI-STAR	0.9297	0.9255	0.0007	0.2440
15x15C04	HI-STAR	0.9311	0.9268	0.0008	0.2435
B15x15C01	HI-TRAC	0.9362	0.9319	0.0008	0.2374
B15x15C01	HI-STAR	0.9361	0.9316	0.0009	0.2385

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
15x15D01	HI-STAR	0.9341	0.9298	0.0008	0.2822
15x15D02	HI-STAR	0.9367	0.9324	0.0008	0.2802
15x15D03	HI-STAR	0.9354	0.9311	0.0008	0.2844
15x15D04	HI-TRAC	0.9354	0.9309	0.0009	0.2963
15x15D04	HI-STAR	0.9339	0.9292	0.0010	0.2958
15x15E01	HI-TRAC	0.9392	0.9349	0.0008	0.2827
15x15E01	HI-STAR	0.9368	0.9325	0.0008	0.2826
15x15F01	HI-STORM	0.3648	0.3614	0.0003	3.03E+04
15x15F01	HI-TRAC	0.9393	0.9347	0.0009	0.2925
15x15F01	HI-STAR	0.9395	0.9350	0.0009	0.2903
15x15G01	HI-TRAC	0.8878	0.8836	0.0007	0.3347
15x15G01	HI-STAR	0.8876	0.8833	0.0008	0.3357
15x15H01	HI-TRAC	0.9333	0.9288	0.0009	0.2353
15x15H01	HI-STAR	0.9337	0.9292	0.0009	0.2349
16x16A01	HI-STORM	0.3447	0.3412	0.0004	3.15E+04
16x16A01	HI-TRAC	0.9273	0.9228	0.0009	0.2710
16x16A01	HI-STAR	0.9287	0.9244	0.0008	0.2704
16x16A02	HI-STAR	0.9263	0.9221	0.0007	0.2702
<del>17x17A01</del>	<del>HI-STAR</del>	<del>0.9368</del>	<del>0.9325</del>	<del>0.0008</del>	<del>0.2131</del>
17x17A02/	HI-STORM	0.3243	0.3210	0.0003	3.23E+04
17x17A02/	HI-TRAC	0.9378	0.9335	0.0008	0.2133
17x17A02/	HI-STAR	0.9368	0.9325	0.0008	0.2131

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
17x17A032	HI-STAR	0.9329	0.9286	0.0008	0.2018
17x17B01	HI-STAR	0.9288	0.9243	0.0009	0.2607
17x17B02	HI-STAR	0.9290	0.9247	0.0008	0.2596
17x17B03	HI-STAR	0.9243	0.9199	0.0008	0.2625
17x17B04	HI-STAR	0.9324	0.9279	0.0009	0.2576
17x17B05	HI-STAR	0.9266	0.9222	0.0008	0.2539
17x17B06	HI-TRAC	0.9318	0.9275	0.0008	0.2570
17x17B06	HI-STAR	0.9311	0.9268	0.0008	0.2593
17x17C01	HI-STAR	0.9293	0.9250	0.0008	0.2595
17x17C02	HI-TRAC	0.9319	0.9274	0.0009	0.2610
17x17C02	HI-STAR	0.9336	0.9293	0.0008	0.2624



Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
6x6A01	HI-STAR	0.7539	0.7498	0.0007	0.2754
6x6A02	HI-STAR	0.7517	0.7476	0.0007	0.2510
6x6A03	HI-STAR	0.7545	0.7501	0.0008	0.2494
6x6A04	HI-STAR	0.7537	0.7494	0.0008	0.2494
6x6A05	HI-STAR	0.7555	0.7512	0.0008	0.2470
6x6A06	HI-STAR	0.7618	0.7576	0.0008	0.2298
6x6A07	HI-STAR	0.7588	0.7550	0.0005	0.2360
6x6A08	HI-STAR	0.7808	0.7766	0.0007	0.2527
B6x6A01	HI-TRAC	0.7732	0.7691	0.0007	0.2458
B6x6A01	HI-STAR	0.7727	0.7685	0.0007	0.2460
B6x6A02	HI-TRAC	0.7785	0.7741	0.0008	0.2411
B6x6A02	HI-STAR	0.7782	0.7738	0.0008	0.2408
B6x6A03	HI-TRAC	0.7886	0.7846	0.0007	0.2311
B6x6A03	HI-STAR	0.7888	0.7846	0.0007	0.2310
6x6B01	HI-STAR	0.7604	0.7563	0.0007	0.2461
6x6B02	HI-STAR	0.7618	0.7577	0.0007	0.2450
6x6B03	HI-STAR	0.7619	0.7578	0.0007	0.2439
6x6B04	HI-STAR	0.7686	0.7644	0.0008	0.2286
6x6B05	HI-STAR	0.7824	0.7785	0.0006	0.2184
B6x6B01	HI-TRAC	0.7833	0.7794	0.0006	0.2181
B6x6B01	HI-STAR	0.7822	0.7783	0.0006	0.2190

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
6x6C01	HI-STORM	0.2759	0.2726	0.0003	1.59E+04
6x6C01	HI-TRAC	0.8024	0.7982	0.0008	0.2135
6x6C01	HI-STAR	0.8021	0.7980	0.0007	0.2139
7x7A01	HI-TRAC	0.7963	0.7922	0.0007	0.2016
7x7A01	HI-STAR	0.7974	0.7932	0.0008	0.2015
7x7B01	HI-STAR	0.9372	0.9330	0.0007	0.3658
7x7B02	HI-STAR	0.9301	0.9260	0.0007	0.3524
7x7B03	HI-STAR	0.9313	0.9271	0.0008	0.3438
7x7B04	HI-STAR	0.9311	0.9270	0.0007	0.3816
7x7B05	HI-STAR	0.9350	0.9306	0.0008	0.3382
7x7B06	HI-STAR	0.9298	0.9260	0.0006	0.3957
B7x7B01	HI-TRAC	0.9367	0.9324	0.0008	0.3899
B7x7B01	HI-STAR	0.9375	0.9332	0.0008	0.3887
B7x7B02	HI-STORM	0.4061	0.4027	0.0003	2.069E+04
B7x7B02	HI-TRAC	0.9385	0.9342	0.0008	0.3952
B7x7B02	HI-STAR	0.9386	0.9344	0.0007	0.3983
8x8A01	HI-TRAC	0.7662	0.7620	0.0008	0.2250
8x8A01	HI-STAR	0.7685	0.7644	0.0007	0.2227
8x8A02	HI-TRAC	0.7690	0.7650	0.0007	0.2163
8x8A02	HI-STAR	0.7697	0.7656	0.0007	0.2158
8x8B01	HI-STAR	0.9310	0.9265	0.0009	0.2935
8x8B02	HI-STAR	0.9227	0.9185	0.0007	0.2993

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
8x8B03	HI-STAR	0.9299	0.9257	0.0008	0.3319
8x8B04	HI-STAR	0.9236	0.9194	0.0008	0.3700
B8x8B01	HI-TRAC	0.9352	0.9310	0.0008	0.3393
B8x8B01	HI-STAR	0.9346	0.9301	0.0009	0.3389
B8x8B02	HI-TRAC	0.9401	0.9359	0.0007	0.3331
B8x8B02	HI-STAR	0.9385	0.9343	0.0008	0.3329
B8x8B03	HI-STORM	0.3934	0.3900	0.0004	1.815E+04
B8x8B03	HI-TRAC	0.9427	0.9385	0.0008	0.3278
B8x8B03	HI-STAR	0.9416	0.9375	0.0007	0.3293
8x8C01	HI-STAR	0.9315	0.9273	0.0007	0.2822
8x8C02	HI-STAR	0.9313	0.9268	0.0009	0.2716
8x8C03	HI-STAR	0.9329	0.9286	0.0008	0.2877
8x8C04	HI-STAR	0.9348	0.9307	0.0007	0.2915
8x8C05	HI-STAR	0.9353	0.9312	0.0007	0.2971
8x8C06	HI-STAR	0.9353	0.9312	0.0007	0.2944
8x8C07	HI-STAR	0.9314	0.9273	0.0007	0.2972
8x8C08	HI-STAR	0.9339	0.9298	0.0007	0.2915
8x8C09	HI-STAR	0.9301	0.9260	0.0007	0.3183
8x8C10	HI-STAR	0.9317	0.9275	0.0008	0.3018
8x8C11	HI-STAR	0.9328	0.9287	0.0007	0.3001
8x8C12	HI-STAR	0.9285	0.9242	0.0008	0.3062
B8x8C01	HI-TRAC	0.9348	0.9305	0.0008	0.3114

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
B8x8C01	HI-STAR	0.9357	0.9313	0.0009	0.3141
B8x8C02	HI-STORM	0.3714	0.3679	0.0004	2.30E+04
B8x8C02	HI-TRAC	0.9402	0.9360	0.0008	0.3072
B8x8C02	HI-STAR	0.9425	0.9384	0.0007	0.3081
B8x8C03	HI-TRAC	0.9429	0.9386	0.0008	0.3045
B8x8C03	HI-STAR	0.9418	0.9375	0.0008	0.3056
8x8D01	HI-STAR	0.9342	0.9302	0.0006	0.2733
8x8D02	HI-STAR	0.9325	0.9284	0.0007	0.2750
8x8D03	HI-STAR	0.9351	0.9309	0.0008	0.2731
8x8D04	HI-STAR	0.9338	0.9296	0.0007	0.2727
8x8D05	HI-STAR	0.9339	0.9294	0.0009	0.2700
8x8D06	HI-STAR	0.9365	0.9324	0.0007	0.2777
8x8D07	HI-STAR	0.9341	0.9297	0.0009	0.2694
8x8D08	HI-STAR	0.9376	0.9332	0.0009	0.2841
B8x8D01	HI-TRAC	0.9408	0.9368	0.0006	0.2773
B8x8D01	HI-STAR	0.9403	0.9363	0.0007	0.2778
8x8E01	HI-TRAC	0.9309	0.9266	0.0008	0.2834
8x8E01	HI-STAR	0.9312	0.9270	0.0008	0.2831
8x8F01	HI-TRAC	0.9396	0.9356	0.0006	0.2255
8x8F01	HI-STAR	0.9411	0.9366	0.0009	0.2264
9x9A01	HI-STAR	0.9353	0.9310	0.0008	0.2875
9x9A02	HI-STAR	0.9388	0.9345	0.0008	0.2228

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
9x9A03	HI-STAR	0.9351	0.9310	0.0007	0.2837
9x9A04	HI-STAR	0.9396	0.9355	0.0007	0.2262
B9x9A01	HI-STORM	0.3365	0.3331	0.0003	1.78E+04
B9x9A01	HI-TRAC	0.9434	0.9392	0.0007	0.2232
B9x9A01	HI-STAR	0.9417	0.9374	0.0008	0.2236
9x9B01	HI-STAR	0.9380	0.9336	0.0008	0.2576
9x9B02	HI-STAR	0.9373	0.9329	0.0009	0.2578
9x9B03	HI-STAR	0.9417	0.9374	0.0008	0.2545
B9x9B01	HI-TRAC	0.9417	0.9376	0.0007	0.2504
B9x9B01	HI-STAR	0.9436	0.9394	0.0008	0.2506
9x9C01	HI-TRAC	0.9377	0.9335	0.0008	0.2697
9x9C01	HI-STAR	0.9395	0.9352	0.0008	0.2698
9x9D01	HI-TRAC	0.9387	0.9343	0.0008	0.2635
9x9D01	HI-STAR	0.9394	0.9350	0.0009	0.2625
9x9E01	HI-STAR	0.9334	0.9293	0.0007	0.2227
9x9E02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9E02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9E02	HI-STAR	0.9401	0.9359	0.0008	0.2065
9x9F01	HI-STAR	0.9307	0.9265	0.0007	0.2899
9x9F02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9F02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9F02	HI-STAR	0.9401	0.9359	0.0008	0.2065

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-68</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
9x9G01	HI-TRAC	0.9307	0.9265	0.0007	0.2193
9x9G01	HI-STAR	0.9309	0.9265	0.0008	0.2191
10x10A01	HI-STAR	0.9377	0.9335	0.0008	0.3170
10x10A02	HI-STAR	0.9426	0.9386	0.0007	0.2159
10x10A03	HI-STAR	0.9396	0.9356	0.0007	0.3169
B10x10A01	HI-STORM	0.3379	0.3345	0.0003	1.74E+04
B10x10A01	HI-TRAC	0.9448	0.9405	0.0008	0.2214
B10x10A01	HI-STAR	0.9457	0.9414	0.0008	0.2212
10x10B01	HI-STAR	0.9384	0.9341	0.0008	0.2881
10x10B02	HI-STAR	0.9416	0.9373	0.0008	0.2333
10x10B03	HI-STAR	0.9375	0.9334	0.0007	0.2856
B10x10B01	HI-TRAC	0.9443	0.9401	0.0007	0.2380
B10x10B01	HI-STAR	0.9436	0.9395	0.0007	0.2366
10x10C01	HI-TRAC	0.9430	0.9387	0.0008	0.2424
10x10C01	HI-STAR	0.9433	0.9392	0.0007	0.2416
10x10D01	HI-TRAC	0.9383	0.9343	0.0007	0.3359
10x10D01	HI-STAR	0.9376	0.9333	0.0008	0.3355
10x10E01	HI-TRAC	0.9157	0.9116	0.0007	0.3301
10x10E01	HI-STAR	0.9185	0.9144	0.0007	0.2936

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24 400PPM SOLUBLE BORON</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.8884	0.8841	0.0008	0.2501
B14x14B01	HI-STAR	0.8900	0.8855	0.0009	0.3173
14x14C03	HI-STAR	0.8950	0.8907	0.0008	0.3410
14x14D01	HI-STAR	0.8518	0.8475	0.0008	0.4395
14x14E02	HI-STAR	0.7132	0.7090	0.0007	0.4377
15x15A01	HI-STAR	0.9119	0.9076	0.0008	0.3363
B15x15B01	HI-STAR	0.9284	0.9241	0.0008	0.3398
B15x15C01	HI-STAR	0.9236	0.9193	0.0008	0.3074
15x15D04	HI-STAR	0.9261	0.9218	0.0008	0.3841
15x15E01	HI-STAR	0.9265	0.9221	0.0008	0.3656
15x15F01	HI-STORM (DRY)	0.4013	0.3978	0.0004	28685
15x15F01	HI-TRAC	0.9301	0.9256	0.0009	0.3790
15x15F01	HI-STAR	0.9314	0.9271	0.0008	0.3791
15x15G01	HI-STAR	0.8939	0.8897	0.0007	0.4392
15x15H01	HI-TRAC	0.9345	0.9301	0.0008	0.3183
15x15H01	HI-STAR	0.9366	0.9320	0.0009	0.3175
16x16A01	HI-STAR	0.8955	0.8912	0.0008	0.3227
17x17A02/	HI-STAR	0.9264	0.9221	0.0008	0.2801
17x17B06	HI-STAR	0.9284	0.9241	0.0008	0.3383
17x17C02	HI-TRAC	0.9296	0.9250	0.0009	0.3447
17x17C02	HI-STAR	0.9294	0.9249	0.0009	0.3433

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24E/MPC-24EF, UNBORATED WATER</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.9380	0.9337	0.0008	0.2277
B14x14B01	HI-STAR	0.9312	0.9269	0.0008	0.2927
14x14C01	HI-STAR	0.9356	0.9311	0.0009	0.3161
14x14D01	HI-STAR	0.8875	0.8830	0.0009	0.4026
14x14E02	HI-STAR	0.7651	0.7610	0.0007	0.3645
15x15A01	HI-STAR	0.9336	0.9292	0.0008	0.2879
B15x15B01	HI-STAR	0.9465	0.9421	0.0008	0.2924
B15x15C01	HI-STAR	0.9462	0.9419	0.0008	0.2631
15x15D04	HI-STAR	0.9440	0.9395	0.0009	0.3316
15x15E01	HI-STAR	0.9455	0.9411	0.0009	0.3178
15x15F01	HI-STORM (DRY)	0.3699	0.3665	0.0004	3.280e+04
15x15F01	HI-TRAC	0.9465	0.9421	0.0009	0.3297
15x15F01	HI-STAR	0.9468	0.9424	0.0008	0.3270
15x15G01	HI-STAR	0.9054	0.9012	0.0007	0.3781
15x15H01	HI-STAR	0.9423	0.9381	0.0008	0.2628
16x16A01	HI-STAR	0.9341	0.9297	0.0009	0.3019
17x17A02/	HI-TRAC	0.9467	0.9425	0.0008	0.2372
17x17A02/	HI-STAR	0.9447	0.9406	0.0007	0.2374
17x17B06	HI-STAR	0.9421	0.9377	0.0008	0.2888
17x17C02	HI-STAR	0.9433	0.9390	0.0008	0.2932



Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-24E/MPC-24EF, 300PPM BORATED WATER</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.8963	0.8921	0.0008	0.2231
B14x14B01	HI-STAR	0.8974	0.8931	0.0008	0.3214
14x14C01	HI-STAR	0.9031	0.8988	0.0008	0.3445
14x14D01	HI-STAR	0.8588	0.8546	0.0007	0.4407
14x14E02	HI-STAR	0.7249	0.7205	0.0008	0.4186
15x15A01	HI-STAR	0.9161	0.9118	0.0008	0.3408
B15x15B01	HI-STAR	0.9321	0.9278	0.0008	0.3447
B15x15C01	HI-STAR	0.9271	0.9227	0.0008	0.3121
15x15D04	HI-STAR	0.9290	0.9246	0.0009	0.3950
15x15E01	HI-STAR	0.9309	0.9265	0.0009	0.3754
15x15F01	HI-STORM (DRY)	0.3897	0.3863	0.0003	3.192E+04
15x15F01	HI-TRAC	0.9333	0.9290	0.0008	0.3900
15x15F01	HI-STAR	0.9332	0.9289	0.0008	0.3861
15x15G01	HI-STAR	0.8972	0.8930	0.0007	0.4473
15x15H01	HI-TRAC	0.9399	0.9356	0.0008	0.3235
15x15H01	HI-STAR	0.9399	0.9357	0.0008	0.3248
16x16A01	HI-STAR	0.9021	0.8977	0.0009	0.3274
17x17A02/	HI-STAR	0.9332	0.9287	0.0009	0.2821
17x17B06	HI-STAR	0.9316	0.9273	0.0008	0.3455
17x17C02	HI-TRAC	0.9320	0.9277	0.0008	0.2819
17x17C02	HI-STAR	0.9312	0.9270	0.0007	0.3530

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-32, 1900 PPM BORATED WATER 4.1% Enrichment, Bounding Cases</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
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
16x16A01	HI-STAR	0.9468	0.9427	0.0007	0.3925
17x17A02/1	HI-STAR	0.9111	0.9072	0.0006	0.4055
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365
17x17C02	HI-TRAC	0.9365	0.9327	0.0006	0.4468
17x17C02	HI-STAR	0.9355	0.9317	0.0006	0.4469

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<b>MPC-32 2600 PPM BORATED WATER, GUIDE TUBES FILLED</b>					
<b>Fuel Assembly Designation</b>	<b>Cask</b>	<b>Maximum <math>k_{eff}</math></b>	<b>Calculated <math>k_{eff}</math></b>	<b>Std. Dev. (1-sigma)</b>	<b>EALF (eV)</b>
14x14A03	HI-STAR	0.8362	0.8324	0.0006	0.4651
B14x14B01	HI-STAR	0.8633	0.8594	0.0006	0.5923
14x14C01	HI-STAR	0.8808	0.8768	0.0007	0.6567
14x14D01	HI-STAR	0.8485	0.8446	0.0006	0.7957
14x14E02	HI-STAR	0.6240	0.6200	0.0006	0.9061
15x15A01	HI-STAR	0.9121	0.9082	0.0006	0.6343
B15x15B01	HI-STAR	0.9286	0.9247	0.0006	0.6613
B15x15C01	HI-STAR	0.9150	0.9110	0.0007	0.5997
15x15D04	HI-STAR	0.9419	0.9379	0.0006	0.7572
15x15E01	HI-STAR	0.9415	0.9376	0.0006	0.7194
15x15F01	HI-STORM (DRY)	0.5142	0.5108	0.0004	1.228E+04
15x15F01	HI-TRAC	0.9463	0.9423	0.0007	0.7409
15x15F01	HI-STAR	0.9465	0.9425	0.0006	0.7421
15x15G01	HI-STAR	0.9109	0.9070	0.0006	0.8486
15x15H01	HI-STAR	0.9301	0.9263	0.0006	0.6257
16x16A01	HI-STAR	0.8868	0.8829	0.0006	0.6105
17x17A02	HI-STAR	0.9145	0.9105	0.0006	0.5382
17x17B06	HI-STAR	0.9358	0.9318	0.0007	0.6500
17x17C02	HI-TRAC	0.9424	0.9385	0.0006	0.6659
17x17C02	HI-STAR	0.9431	0.9391	0.0006	0.6628

Table 6.C.1 (continued)  
 CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
 AND BASKET CONFIGURATIONS

<del>MPC-32, 2600 PPM BORATED WATER, GUIDE TUBES VOIDED</del> 5.0% Enrichment, Bounding Cases					
Fuel Assembly Designation	Cask	Maximum $k_{eff}$	Calculated $k_{eff}$	Std. Dev. (1-sigma)	EALF (eV)
14x14A03	HI-STAR	0.9000	0.8959	0.0007	0.4651
B14x14B01	HI-STAR	0.9214	0.9175	0.0006	0.6009
14x14C01	HI-STAR	0.9480	0.9440	0.0006	0.6431
14x14D01	HI-STAR	0.9050	0.9009	0.0007	0.7276
14x14E02	HI-STAR	0.7415	0.7375	0.0006	0.9226
15x15A01	HI-STAR	0.9230	0.9189	0.0007	0.7143
B15x15B01	HI-STAR	0.9429	0.9390	0.0006	0.7234
B15x15C01	HI-STAR	0.9307	0.9268	0.0006	0.6439
15x15D04	HI-STAR	0.9466	0.9425	0.0007	0.7525
15x15E01	HI-STAR	0.9434	0.9394	0.0007	0.7215
15x15F01	HI-STORM (DRY)	0.5142	0.5108	0.0004	1.228E+04
15x15F01	HI-TRAC	0.9470	0.9431	0.0006	0.7456
15x15F01	HI-STAR	0.9483	0.9443	0.0007	0.7426
15x15G01	HI-STAR	0.9251	0.9212	0.0006	0.9303
15x15H01	HI-STAR	0.9333	0.9292	0.0007	0.7015
16x16A01	HI-STAR	0.9474	0.9434	0.0006	0.5936
17x17A02/1	HI-STAR	0.9161	0.9122	0.0006	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

Table 6.C.1 (continued)  
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES  
AND BASKET CONFIGURATIONS

Note: Maximum  $k_{\text{eff}} = \text{Calculated } k_{\text{eff}} + K_c \times \sigma_c + \text{Bias} + \sigma_B$   
where:

$$\begin{aligned} K_c &= 2.0 \\ \sigma_c &= \text{Std. Dev. (1-sigma)} \\ \text{Bias} &= 0.0021 \\ \sigma_B &= 0.0006 \end{aligned}$$

See Subsection 6.4.3 for further explanation.

## APPENDIX 6.D: SAMPLE INPUT FILES

(Total number of pages in this appendix : 35)

File Description	Starting Page
MCNP4a input file for MPC-24 in HI-TRAC	Appendix 6.D-2
MCNP4a input file for MPC-68 in HI-TRAC	Appendix 6.D-13
MCNP4a input file for MPC-24 in HI-STORM	Appendix 6.D-19
MCNP4a input file for MPC-68 in HI-STORM	Appendix 6.D-30

HI-TRAC Transfer Cask containing MPC24, 17x17 assembly @ 4.0 wt% Enrich.

```
c
c
c
c MPC-24/24E cell configuration
c
c HI-TRAC with active length 150 inch
c
c Cask Input Preprocessor
c cskinp 17a 17a mpc24n mpc24n hitrac trac150 4.0 4tf7a45 pure
c ----- cpp\17a.bat
c   added 17a.ce
c   added 17a.su
c   added 17a.sp
c ----- cpp\mpc24n.bat
c   added mpc24n.co
c   added mpc24n.ce
c   added mpc24n.su
c   added mpc24n.sp
c ----- cpp\hitrac.bat
c   added trac150.co
c   added hitrac.ce
c   added trac150.su
c   added trac150.sp
c end of comments
c
c start of cells
c
c 17x17a
c
c number of cells: 6
c cell numbers:      1 to 7
c univers numbers:   1 to 3
c surface numbers:   1 to 9
c
c number of cells: 1
1   1 -10.522   -1 u=2      $ fuel
2   4 -1.0      1 -2 u=2      $ gap
3   3 -6.55     2 -3 u=2      $ Zr Clad
4   4 -1.0      3 u=2      $ water in fuel region
5   4 -1.0   -4:5   u=3      $ water in guide tubes
6   3 -6.55     4 -5   u=3      $ guide tubes
7   4 -1.0   -6   +7   -8   +9   u=1 lat=1
fill= -9:9   -9:9   0:0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 2 2 1
1 2 2 2 3 2 2 2 2 2 2 2 2 2 3 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
```

```

1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
c
c MPC-24
c
c number of cells: 102
c cell numbers : 400 to 699
c universe numbers : 4 to 9
c surface numbers : 400 to 699
c
c Right Side
c
408 0 -410 411 -412 413 u=4 fill=1 (1)
409 5 -7.84 410 -424 413 -426 u=4
410 4 -1.0 424 -428 448 -445 u=4
411 7 -2.7 428 -528 448 -445 u=4
412 6 -2.66 528 -532 448 -445 u=4
413 7 -2.7 532 -432 448 -445 u=4
414 4 -1.0 432 -436 448 -445 u=4
415 5 -7.84 436 -440 448 -445 u=4
416 4 -1.0 440 413 u=4
417 4 -1.0 424 -440 413 -447 u=4
418 4 -1.0 424 -440 446 u=4
419 5 -7.84 424 -440 447 -448 u=4
420 5 -7.84 424 -440 445 -446 u=4
c
c Left Side
c
421 5 -7.84 425 -411 413 u=4
422 4 -1.0 429 -425 448 -445 u=4
423 7 -2.7 529 -429 448 -445 u=4
424 6 -2.66 533 -529 448 -445 u=4
425 7 -2.7 433 -533 448 -445 u=4
426 4 -1.0 437 -433 448 -445 u=4
427 5 -7.84 441 -437 448 -445 u=4
428 4 -1.0 -441 413 u=4
429 4 -1.0 441 -425 413 -447 u=4
430 4 -1.0 441 -425 446 u=4
431 5 -7.84 441 -425 447 -448 u=4
432 5 -7.84 441 -425 445 -446 u=4
c
c Top
c
433 5 -7.84 411 -410 412 -426 u=4
434 4 -1.0 451 -452 426 -430 u=4
435 7 -2.7 451 -452 430 -530 u=4
436 6 -2.66 451 -452 530 -534 u=4
437 7 -2.7 451 -452 534 -434 u=4
438 4 -1.0 451 -452 434 -438 u=4
439 5 -7.84 451 -452 438 -442 u=4
440 4 -1.0 411 -424 442 u=4
441 4 -1.0 411 -450 426 -442 u=4
442 4 -1.0 453 -424 426 -442 u=4
443 5 -7.84 450 -451 426 -442 u=4
444 5 -7.84 452 -453 426 -442 u=4
c
c Bottom
c
445 5 -7.84 427 -413 u=4
446 4 -1.0 451 -452 431 -427 u=4
447 7 -2.7 451 -452 531 -431 u=4
448 6 -2.66 451 -452 535 -531 u=4
449 7 -2.7 451 -452 435 -535 u=4

```



```

450  4 -1.0      451 -452      439 -435      u=4
451  5 -7.84    451 -452      443 -439      u=4
452  4 -1.0      411          -443      u=4
453  4 -1.0      411 -450      443 -427      u=4
454  4 -1.0      453          443 -427      u=4
455  5 -7.84    450 -451      443 -427      u=4
456  5 -7.84    452 -453      443 -427      u=4
457  5 -7.84    425 -411      -427      u=4
458  4 -1.0      -425          -427      u=4
c
c   TYPE B CELL - Short Boral on top and right
c
c   Right Side
c
459  0          -410  411  -412  413      u=5 fill=1 (1)
460  5 -7.84    410 -424      413 -426      u=5
470  4 -1.0      424 -428      548 -545      u=5
471  7 -2.7      428 -528      548 -545      u=5
472  6 -2.66    528 -532      548 -545      u=5
473  7 -2.7      532 -432      548 -545      u=5
474  4 -1.0      432 -436      548 -545      u=5
475  5 -7.84    436 -440      548 -545      u=5
476  4 -1.0      440          413          u=5
477  4 -1.0      424 -440      413 -547      u=5
478  4 -1.0      424 -440      546          u=5
479  5 -7.84    424 -440      547 -548      u=5
480  5 -7.84    424 -440      545 -546      u=5
c
c   Left Side
c
481  5 -7.84    425 -411      413          u=5
482  4 -1.0      429 -425      448 -445      u=5
483  7 -2.7      529 -429      448 -445      u=5
484  6 -2.66    533 -529      448 -445      u=5
485  7 -2.7      433 -533      448 -445      u=5
486  4 -1.0      437 -433      448 -445      u=5
487  5 -7.84    441 -437      448 -445      u=5
488  4 -1.0      -441          413          u=5
489  4 -1.0      441 -425      413 -447      u=5
490  4 -1.0      441 -425      446          u=5
491  5 -7.84    441 -425      447 -448      u=5
492  5 -7.84    441 -425      445 -446      u=5
c
c   Top
c
493  5 -7.84    411 -410      412 -426      u=5
494  4 -1.0      551 -552      426 -430      u=5
495  7 -2.7      551 -552      430 -530      u=5
496  6 -2.66    551 -552      530 -534      u=5
497  7 -2.7      551 -552      534 -434      u=5
498  4 -1.0      551 -552      434 -438      u=5
499  5 -7.84    551 -552      438 -442      u=5
500  4 -1.0      411 -424      442          u=5
501  4 -1.0      411 -550      426 -442      u=5
502  4 -1.0      553 -424      426 -442      u=5
503  5 -7.84    550 -551      426 -442      u=5
504  5 -7.84    552 -553      426 -442      u=5
c
c   Bottom
c
505  5 -7.84    427          -413      u=5
506  4 -1.0      451 -452      431 -427      u=5

```

```

507  7  -2.7    451  -452  531  -431    u=5
508  6  -2.66   451  -452  535  -531    u=5
509  7  -2.7    451  -452  435  -535    u=5
510  4  -1.0    451  -452  439  -435    u=5
511  5  -7.84   451  -452  443  -439    u=5
512  4  -1.0    411                -443    u=5
513  4  -1.0    411  -450  443  -427    u=5
514  4  -1.0    453                443  -427    u=5
515  5  -7.84   450  -451  443  -427    u=5
516  5  -7.84   452  -453  443  -427    u=5
517  5  -7.84   425  -411                -427    u=5
518  4  -1.0                -425        -427    u=5

```

c  
c  
c

c TYPE D CELL - Short Boral on left and bottom, different cell ID

c

c number of cells: 51

c

c Right Side

c

```

1570  0          -1410  1411  -1412  1413    u=17 fill=1 (1)
1571  5  -7.84   1410  -1424  1413  -1426    u=17
1572  4  -1.0    1424  -1428  1448  -1445    u=17
1573  7  -2.7    1428  -1528  1448  -1445    u=17
1574  6  -2.66   1528  -1532  1448  -1445    u=17
1575  7  -2.7    1532  -1432  1448  -1445    u=17
1576  4  -1.0    1432  -1436  1448  -1445    u=17
1577  5  -7.84   1436  -1440  1448  -1445    u=17
1578  4  -1.0    1440                1413    u=17
1579  4  -1.0    1424  -1440  1413  -1447    u=17
1580  4  -1.0    1424  -1440  1446                u=17
1581  5  -7.84   1424  -1440  1447  -1448    u=17
1582  5  -7.84   1424  -1440  1445  -1446    u=17

```

c

c Left Side

c

```

1583  5  -7.84   1425  -1411  1413                u=17
1584  4  -1.0    1429  -1425  1548  -1545    u=17
1585  7  -2.7    1529  -1429  1548  -1545    u=17
1586  6  -2.66   1533  -1529  1548  -1545    u=17
1587  7  -2.7    1433  -1533  1548  -1545    u=17
1588  4  -1.0    1437  -1433  1548  -1545    u=17
1589  5  -7.84   1441  -1437  1548  -1545    u=17
1590  4  -1.0                -1441  1413    u=17
1591  4  -1.0    1441  -1425  1413  -1547    u=17
1592  4  -1.0    1441  -1425  1546                u=17
1593  5  -7.84   1441  -1425  1547  -1548    u=17
1594  5  -7.84   1441  -1425  1545  -1546    u=17

```

c

c Top

c

```

1595  5  -7.84   1411  -1410  1412  -1426    u=17
1596  4  -1.0    1451  -1452  1426  -1430    u=17
1597  7  -2.7    1451  -1452  1430  -1530    u=17
1598  6  -2.66   1451  -1452  1530  -1534    u=17
1599  7  -2.7    1451  -1452  1534  -1434    u=17
1600  4  -1.0    1451  -1452  1434  -1438    u=17
1601  5  -7.84   1451  -1452  1438  -1442    u=17
1602  4  -1.0    1411  -1424  1442                u=17
1603  4  -1.0    1411  -1450  1426  -1442    u=17
1604  4  -1.0    1453  -1424  1426  -1442    u=17

```

```

1605 5 -7.84 1450 -1451 1426 -1442 u=17
1606 5 -7.84 1452 -1453 1426 -1442 u=17
c
c Bottom
c
1607 5 -7.84 1427 -1413 u=17
1608 4 -1.0 1551 -1552 1431 -1427 u=17
1609 7 -2.7 1551 -1552 1531 -1431 u=17
1610 6 -2.66 1551 -1552 1535 -1531 u=17
1611 7 -2.7 1551 -1552 1435 -1535 u=17
1612 4 -1.0 1551 -1552 1439 -1435 u=17
1613 5 -7.84 1551 -1552 1443 -1439 u=17
1614 4 -1.0 1411 -1443 u=17
1615 4 -1.0 1411 -1550 1443 -1427 u=17
1616 4 -1.0 1553 1443 -1427 u=17
1617 5 -7.84 1550 -1551 1443 -1427 u=17
1618 5 -7.84 1552 -1553 1443 -1427 u=17
1619 5 -7.84 1425 -1411 -1427 u=17
1620 4 -1.0 -1425 -1427 u=17

```

```

c
c number of cells: 29
c
c empty cell no borral, no top
c

```

```

751 4 -1.0 -410 411 -412 413 u=14
752 5 -7.84 410 -424 413 -426 u=14
753 5 -7.84 425 -411 413 u=14
754 4 -1.0 411 -410 412 -426 u=14
755 5 -7.84 427 -413 u=14
756 5 -7.84 425 -411 -427 u=14
757 4 -1.0 411 426 u=14
758 4 -1.0 411 -427 u=14
759 4 -1.0 -425 413 u=14
760 4 -1.0 424 413 -426 u=14
761 4 -1.0 -425 -427 u=14

```

```

c
c
701 5 -7.84 701 -702 711 -713 u=9 $ steel post
702 5 -7.84 702 -703 711 -712 u=9 $ steel post
c
711 0 701 -705 711 -715 (702:713) (703:712)
fill=4 (13.8506 13.8506 0) u=9
712 0 704 (-706:-716) (705:715) -717 -710
fill=4 (17.9489 41.5518 0 0 1 0 -1 0 0 0 0 1) u=9
713 0 (705:715) -707 714 (-706:-716) 710
fill=4 (41.5518 17.9489 0 0 -1 0 1 0 0 0 0 1) u=9
714 0 701 -705 717 -719
fill=5 (13.8506 69.253 0) u=9
715 0 707 -709 711 -715
fill=5 (69.253 13.8506 0) u=9
716 0 706 -708 716 -718
fill=17 (45.6501 45.6501 0 -1 0 0 0 -1 0 0 0 1) u=9
717 0 705 -706 717 -719
fill=14 (41.5518 69.253 0) u=9
718 0 707 -709 715 -716
fill=14 (69.253 41.5518 0 0 1 0 1 0 0 0 0 1) u=9
719 0 701 -704 715 -717
fill=14 (-9.75233 41.5518 0 -1 0 0 0 1 0 0 0 1) u=9
720 0 705 -707 711 -714
fill=14 (41.5518 -9.75233 0 0 -1 0 1 0 0 0 0 1) u=9
721 4 -1.0 (706:719) (708:718) (709:716) u=9

```

```

c
c
c
731  4 -1.0      720 721  fill=9 (0 0 0) u=19
732  4 -1.0     -720 721  fill=9 (0 0 0
      -1 0 0 0 1 0 0 0 1) u=19
733  4 -1.0      720 -721 fill=9 (0 0 0
      1 0 0 0 -1 0 0 0 1) u=19
734  4 -1.0     -720 -721 fill=9 (0 0 0
      -1 0 0 0 -1 0 0 0 1) u=19
c
673  0          -41          39 -40  fill=19
c
c number of cells: 20
374  4 -1.0     -41          300 -39  $ Water below Fuel (4 in.)
375  5 -7.84   -309         302 -300  $ MPC Steel below Fuel (2.5 in.)
376  5 -7.84   -315         320 -302  $ Transfer Cask Steel (2.0 in.)
377  30 -11.34 -315         321 -320  $ Transfer Cask Lead (2.5 in.)
378  5 -7.84   -315         322 -321  $ Transfer Cask Steel (1.0 in.)
c
379  4 -1.0     -41          40 -301  $ Water above Fuel (6 in.)
380  5 -7.84   -309         301 -303  $ MPC Steel above Fuel (9.5 - 0.06 in)
381  4 -1.0     -309         303 -330  $ Water (1.5 in.)
382  5 -7.84   -315         330 -331  $ Transfer Cask Steel (0.75 in.)
383  31 -1.61  -315         331 -332  $ Transfer Cask Neutron Shield (3.25 in.)
384  5 -7.84   -315         332 -333  $ Transfer Cask Steel (0.5 in.)
c
390  5 -7.84    41 -309  300 -301  $ Radial Steel - MPC shell
391  4 -1.00    309 -310  302 -330  $ Radial Water
392  5 -7.84    310 -311  302 -330  $ Radial Steel - inner shell of Trnsfr Cask
393  30 -11.34  311 -312  302 -330  $ Radial Lead - Transfer Cask lead
394  5 -7.84    312 -313  302 -330  $ Radial Steel - outer shell of Trnsfr Cask
395  4 -1.00    313 -314  302 -330  $ Radial Water - Water Jacket
396  5 -7.84    314 -315  302 -330  $ Radial Steel - outer shell of Water
Jacket
c
300  4 -1.00    340 -341 -345 (315 :-322: 333) $ outer water reflector
301  0          345 :-340: 341 $ outside world
c end of cells
c --- empty line

c --- empty line
c start of surfaces
1   cz          0.3922  $ fuel
2   cz          0.4001  $ clad ID
3   cz          0.4572  $ clad OD
4   cz          0.5613  $ guide ID
5   cz          0.6020  $ guide OD
6   px          0.6299  $ pin pitch
7   px         -0.6299
8   py          0.6299
9   py         -0.6299
c
c
c cell-id      8.98
c cell-pitch   10.906
c wall-thkns   5/16
c angle-thkns  5/16
c boral-gap    0.0035
c boral-gap-o  0.0035
c boral-thkns  0.075
c boral-clad   0.01

```

```

c sheathing      0.0235
c boral-wide     7.5
c boral-narrow   6.25
c
c gap size       1.09
c basket-od      67.335
c
410  px          11.40460 $x 8.98/2
411  px          -11.40460 $x {410} *-1
412  py          11.40460 $x {410}
413  py          -11.40460 $x {411}
416  px          13.85062 $x (10.906 + 5/16 - 5/16) /2
417  px          -13.85062 $x -10.906 + {416}
418  py          13.85062 $x {416}
419  py          -13.85062 $x {417}
424  px          12.19835 $x {410} + 5/16      $ angle
425  px          -12.19835 $x {411} - 5/16     $ box wall
426  py          12.19835 $x {412} + 5/16
427  py          -12.19835 $x {413} - 5/16
428  px          12.20724 $x {424} + 0.0035    $ wall to boral gap
429  px          -12.20724 $x {425} - 0.0035
430  py          12.20724 $x {426} + 0.0035
431  py          -12.20724 $x {427} - 0.0035
432  px          12.39774 $x {428} + 0.075     $ boral
433  px          -12.39774 $x {429} - 0.075
434  py          12.39774 $x {430} + 0.075
435  py          -12.39774 $x {431} - 0.075
436  px          12.40663 $x {432} + 0.0035    $ boral to sheathing gap
437  px          -12.40663 $x {433} - 0.0035
438  py          12.40663 $x {434} + 0.0035
439  py          -12.40663 $x {435} - 0.0035
440  px          12.46632 $x {436} + 0.0235    $ sheathing
441  px          -12.46632 $x {437} - 0.0235
442  py          12.46632 $x {438} + 0.0235
443  py          -12.46632 $x {439} - 0.0235
445  py          9.52500 $x 7.5/2
446  py          9.58469 $x {445} + 0.0235    $ sheathing
447  py          -9.58469 $x {446} *-1
448  py          -9.52500 $x {445} *-1
450  px          -9.58469 $x {447}
451  px          -9.52500 $x {448}
452  px          9.52500 $x {445}
453  px          9.58469 $x {446}
528  px          12.23264 $x {428} + 0.01     $ Aluminum on the outside of boral
529  px          -12.23264 $x {429} - 0.01
530  py          12.23264 $x {430} + 0.01
531  py          -12.23264 $x {431} - 0.01
532  px          12.37234 $x {432} - 0.01
533  px          -12.37234 $x {433} + 0.01
534  py          12.37234 $x {434} - 0.01
535  py          -12.37234 $x {435} + 0.01
545  py          7.93750 $x 6.25/2
546  py          7.99719 $x {545} + 0.0235    $ sheathing
547  py          -7.99719 $x {546} *-1
548  py          -7.93750 $x {545} *-1
550  px          -7.99719 $x {547}
551  px          -7.93750 $x {548}
552  px          7.93750 $x {545}
553  px          7.99719 $x {546}
c
c cell-id-2      8.98
c gap-o          1.09

```

```

c
701 px -5.0
702 px 1.90627 $x (10.906 - 8.98)/2 - 5/16 + 0.1
703 px 3.45694 $x 2.722/2
704 px 4.09829 $x 10.906 - 8.98 - 5/16
705 px 27.70124 $x 10.906
706 px 31.79953 $x 2 * 10.906 - (8.98+8.98)/2 - 5/16
707 px 55.40248 $x 2 * 10.906
708 px 59.50077 $x {707} + {704}
709 px 83.10372 $x 3 * 10.906
710 p 1 -1 0 0.1 $ diagonal x=y, offset by 0.1 to avoid intersecting
corners
711 py -4.99999 $x {701}
712 py 1.90627 $x {702}
713 py 3.45694 $x {703}
714 py 4.09829 $x {704}
715 py 27.70124 $x {705}
716 py 31.79953 $x {706}
717 py 55.40248 $x {707}
718 py 59.50077 $x {708}
719 py 83.10372 $x {709}
720 px 0.0
721 py 0.0
1410 px 11.40460 $x 8.98/2
1411 px -11.40460 $x {1410} *-1
1412 py 11.40460 $x {1410}
1413 py -11.40460 $x {1411}
1424 px 12.19835 $x {1410} + 5/16 $ angle
1425 px -12.19835 $x {1411} - 5/16 $ box wall
1426 py 12.19835 $x {1412} + 5/16
1427 py -12.19835 $x {1413} - 5/16
1428 px 12.20724 $x {1424} + 0.0035 $ wall to boral gap
1429 px -12.20724 $x {1425} - 0.0035
1430 py 12.20724 $x {1426} + 0.0035
1431 py -12.20724 $x {1427} - 0.0035
1432 px 12.39774 $x {1428} + 0.075 $ boral
1433 px -12.39774 $x {1429} - 0.075
1434 py 12.39774 $x {1430} + 0.075
1435 py -12.39774 $x {1431} - 0.075
1436 px 12.40663 $x {1432} + 0.0035 $ boral to sheathing gap
1437 px -12.40663 $x {1433} - 0.0035
1438 py 12.40663 $x {1434} + 0.0035
1439 py -12.40663 $x {1435} - 0.0035
1440 px 12.46632 $x {1436} + 0.0235 $ sheathing
1441 px -12.46632 $x {1437} - 0.0235
1442 py 12.46632 $x {1438} + 0.0235
1443 py -12.46632 $x {1439} - 0.0235
1445 py 9.52500 $x 7.5/2
1446 py 9.58469 $x {1445} + 0.0235 $ sheathing
1447 py -9.58469 $x {1446} *-1
1448 py -9.52500 $x {1445} *-1
1450 px -9.58469 $x {1447}
1451 px -9.52500 $x {1448}
1452 px 9.52500 $x {1445}
1453 px 9.58469 $x {1446}
1528 px 12.23264 $x {1428} + 0.01 $ Aluminum on the outside of boral
1529 px -12.23264 $x {1429} - 0.01
1530 py 12.23264 $x {1430} + 0.01
1531 py -12.23264 $x {1431} - 0.01
1532 px 12.37234 $x {1432} - 0.01
1533 px -12.37234 $x {1433} + 0.01
1534 py 12.37234 $x {1434} - 0.01

```

```

1535 py -12.37234 $x {1435} + 0.01
1545 py 7.93750 $x 6.25/2
1546 py 7.99719 $x {1545} + 0.0235 $ sheathing
1547 py -7.99719 $x {1546} *-1
1548 py -7.93750 $x {1545} *-1
1550 px -7.99719 $x {1547}
1551 px -7.93750 $x {1548}
1552 px 7.93750 $x {1545}
1553 px 7.99719 $x {1546}
39 pz 0.0 $ bottom of active fuel assembly
40 pz 381.0 $ top of active fuel assembly
41 cz 85.57 $ MPC
300 pz -10.16 $ lower water thkness = 4 in.
301 pz 396.24 $ upper water thkness = 6 in.
302 pz -16.51 $ thkness of MPC baseplate = 2.5 in.
303 pz 420.22 $ thkness of MPC lid = 9.5 -0.06 in.
309 cz 86.84 $ I.D. = 68.375 in.
310 cz 87.31 $ I.D. = 68.75 in.
311 cz 89.22 $ I.D. = 70.25 in.
312 cz 100.65 $ I.D. = 79.25 in.
313 cz 103.19 $ I.D. = 81.25 in.
314 cz 116.80 $ I.D. = 91.97 in.
315 cz 118.07 $ I.D. = 92.972 in.
320 pz -21.59 $ thkness steel - 2.0 in.
321 pz -27.94 $ thkness lead - 2.5 in.
322 pz -30.48 $ thkness steel - 1.0 in.
330 pz 424.03 $ thkness water - 1.5 in.
331 pz 425.93 $ thkness steel - 0.75 in.
332 pz 434.19 $ thkness neutron shield - 3.25 in.
333 pz 435.46 $ thkness steel - 0.5 in.
c
*340 pz -60.48 $ lower boundary
*341 pz 465.46 $ upper boundary
*345 cz 148.07 $ outer radial boundary
c end of surfaces
c --- empty line

c --- empty line
tr1 0 0 0
kcode 10000 .94 20 120
dbcn 7j 1e7
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
sp3 0 1
c
si4 s 13 14
12 13 14 15
11 12 13 14 15 16
11 12 13 14 15 16
12 13 14 15
13 14

sp4 1 23r
c
ds5 s 26 26
25 25 25 25
24 24 24 24 24 24
23 23 23 23 23 23
22 22 22 22
21 21
c

```

```

si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639
si14  2.86639  24.50719
si15 30.23997  51.88077
si16 57.61355  79.25435
c
si21 -79.25435 -57.61355
si22 -51.88077 -30.23997
si23 -24.50719 -2.86639
si24  2.86639  24.50719
si25 30.23997  51.88077
si26 57.61355  79.25435
c
sp11 0 1
sp12 0 1
sp13 0 1
sp14 0 1
sp15 0 1
sp16 0 1
sp21 0 1
sp22 0 1
sp23 0 1
sp24 0 1
sp25 0 1
sp26 0 1
c
m3          40000.56c  1.          $ Zr Clad
m4          1001.50c  0.6667         $ Water
           8016.50c  0.3333
m5          24000.50c  0.01761        $ Steel
           25055.50c  0.001761
           26000.55c  0.05977
           28000.50c  0.008239
m6          5010.50c  -0.054427       $ Boral Central Section @ 0.02 g/cmsq
           5011.50c  -0.241373
           13027.50c -0.6222
           6000.50c  -0.0821
m7          13027.50c  1.0
mt4          lwtr.01t
prdmp       j  -120   j   2
fm4         1000   1   -6
f4:n        1
sd4         1000
e4          1.000E-11  1.000E-10  5.000E-10  7.500E-10  1.000E-09  1.200E-09
           1.500E-09  2.000E-09  2.500E-09  3.000E-09
           4.700E-09  5.000E-09  7.500E-09  1.000E-08  2.530E-08
           3.000E-08  4.000E-08  5.000E-08  6.000E-08  7.000E-08
           8.000E-08  9.000E-08  1.000E-07  1.250E-07  1.500E-07
           1.750E-07  2.000E-07  2.250E-07  2.500E-07  2.750E-07
           3.000E-07  3.250E-07  3.500E-07  3.750E-07  4.000E-07
           4.500E-07  5.000E-07  5.500E-07  6.000E-07  6.250E-07
           6.500E-07  7.000E-07  7.500E-07  8.000E-07  8.500E-07
           9.000E-07  9.250E-07  9.500E-07  9.750E-07  1.000E-06
           1.010E-06  1.020E-06  1.030E-06  1.040E-06  1.050E-06
           1.060E-06  1.070E-06  1.080E-06  1.090E-06  1.100E-06
           1.110E-06  1.120E-06  1.130E-06  1.140E-06  1.150E-06
           1.175E-06  1.200E-06  1.225E-06  1.250E-06  1.300E-06
           1.350E-06  1.400E-06  1.450E-06  1.500E-06  1.590E-06
           1.680E-06  1.770E-06  1.860E-06  1.940E-06  2.000E-06
           2.120E-06  2.210E-06  2.300E-06  2.380E-06  2.470E-06
           2.570E-06  2.670E-06  2.770E-06  2.870E-06  2.970E-06

```



3.000E-06	3.050E-06	3.150E-06	3.500E-06	3.730E-06
4.000E-06	4.750E-06	5.000E-06	5.400E-06	6.000E-06
6.250E-06	6.500E-06	6.750E-06	7.000E-06	7.150E-06
8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05
1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05
1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05
2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05
3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05
3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05
4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05
4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05
6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05
8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04
1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04
2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04
5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03
1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03
2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03
8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02
3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02
7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	

si3 h 0 381.00

m30 82000.50c 1.0 \$ Lead

m31 6000.50c -27.660 \$ Neutron Shield Holtite-A (NS-4-FR)

1001.50c -5.920

13027.50c -21.285

7014.50c -1.98

8016.50c -42.372

5010.50c -0.141

5011.50c -0.642

imp:n 1 207r 0

c fuel enrichment 4.0 %

m1 92235.50c -0.03526

92238.50c -0.84624

8016.50c -0.11850

c end of file

c

HI-TRAC Transfer Cask containing MPC68, 08x08 assembly @ 4.2 wt% Enrich.

c reflected w/60cm of water, 0.0279 g/cmsq B-10 in Boral

c

c

1 1 -10.522 -1 u=2 \$ fuel  
2 4 -1.0 1 -2 u=2 \$ gap  
3 3 -6.55 2 -3 u=2 \$ Zr Clad  
4 4 -1.0 3 u=2 \$ water in fuel region  
5 4 -1.0 -4:5 u=3 \$ water in guide tubes  
6 4 -1.00 4 -5 u=3 \$ guide tubes  
7 4 -1.0 -6 +7 -8 +9 u=1 lat=1

fill= -5:4 -5:4 0:0

1 1 1 1 1 1 1 1 1  
1 2 2 2 2 2 2 2 1  
1 2 2 2 2 2 2 2 1  
1 2 2 2 2 2 2 2 1  
1 2 2 2 3 2 2 2 1  
1 2 2 2 3 2 2 2 1  
1 2 2 2 2 2 2 2 1  
1 2 2 2 2 2 2 2 1  
1 2 2 2 2 2 2 2 1  
1 1 1 1 1 1 1 1 1

c

C BOX TYPE R

c

8 0 -10 11 -12 13 u=4 fill=1 (0.8128 0.8128 0)  
9 3 -6.55 60 -61 62 -63 #8 u=4 \$ Zr flow channel  
10 4 -1. 64 -65 66 -67 #8 #9 u=4 \$ water  
11 5 -7.84 20 -23 67 -14 u=4 \$ 0.075" STEEL  
12 4 -1. 20 -23 14 -15 u=4 \$ WATER POCKET  
13 7 -2.7 20 -23 15 -16 u=4 \$ Al CLAD  
14 6 -2.66 20 -23 16 -17 u=4 \$ BORAL Absorber  
15 7 -2.7 20 -23 17 -18 u=4 \$ Al Clad  
16 4 -1. 20 -23 18 -118 u=4 \$ Water  
17 5 -7.84 118:-129:65:-66 u=4 \$ Steel  
18 4 -1. 64 -21 67 -118 u=4 \$ Water  
19 4 -1. 24 -65 67 -118 u=4 \$ water  
20 5 -7.84 21 -20 67 -118 u=4 \$ Steel  
21 5 -7.84 23 -24 67 -118 u=4 \$ Steel  
22 4 -1. 129 -64 33 -118 u=4 \$ Water

c

23 5 -7.84 25 -64 30 -31 u=4 \$ Steel  
24 4 -1. 26 -25 30 -31 u=4 \$ Water  
25 7 -2.7 27 -26 30 -31 u=4 \$ Al clad  
26 6 -2.66 28 -27 30 -31 u=4 \$ Boral  
27 7 -2.7 29 -28 30 -31 u=4 \$ Al clad  
28 4 -1. 129 -29 30 -31 u=4 \$ water  
29 5 -7.84 129 -64 32 -30 u=4 \$ Steel ends  
30 5 -7.84 129 -64 31 -33 u=4 \$ Steel ends  
31 4 -1. 129 -64 66 -32 u=4 \$ Water

c

c Type A box - Boral only on left side

c

32 0 -10 11 -12 13 u=6 fill=1 (0.8128 0.8128 0)  
33 3 -6.55 60 -61 62 -63 #8 u=6 \$ Zr flow channel  
34 4 -1. 64 -65 66 -118 #8 #9 u=6 \$ water  
35 5 -7.84 118:-129:65:-66 u=6 \$ Steel  
36 4 -1. 129 -64 67 -118 u=6 \$ Water  
37 5 -7.84 25 -64 30 -31 u=6 \$ Steel  
38 4 -1. 26 -25 30 -31 u=6 \$ Water  
39 7 -2.7 27 -26 30 -31 u=6 \$ Al clad

```

40 6 -2.66 28 -27 30 -31 u=6 $ Boral
41 7 -2.7 29 -28 30 -31 u=6 $ Al clad
42 4 -1. 129 -29 30 -31 u=6 $ water
43 4 -1. 129 -64 33 -67 u=6 $ Water
44 5 -7.84 129 -64 32 -30 u=6 $ Steel ends
45 5 -7.84 129 -64 31 -33 u=6 $ Steel ends
46 4 -1. 129 -64 66 -32 u=6 $ Water
c
c Type B box - Boral on Top only
c
47 0 -10 11 -12 13 u=7 fill=1 (0.8128 0.8128 0)
48 3 -6.55 60 -61 62 -63 #8 u=7 $ Zr flow channel
49 4 -1. 64 -65 66 -67 #8 #9 u=7 $ water
50 5 -7.84 20 -23 67 -14 u=7 $ 0.075" STEEL
51 4 -1. 20 -23 14 -15 u=7 $ WATER POCKET
52 7 -2.7 20 -23 15 -16 u=7 $ Al CLAD
53 6 -2.66 20 -23 16 -17 u=7 $ BORAL Absorber
54 7 -2.7 20 -23 17 -18 u=7 $ water
55 4 -1. 20 -23 18 -118 u=7 $ Water
56 5 -7.84 118:-129:65:-66 u=7 $ Steel
57 4 -1. 64 -21 67 -118 u=7 $ Water
58 4 -1. 24 -65 67 -118 u=7 $ water
59 5 -7.84 21 -20 67 -118 u=7 $ Steel
60 5 -7.84 23 -24 67 -118 u=7 $ Steel
61 4 -1. 129 -64 66 -118 u=7 $ Water
c
c Type E box - No Boral Panels
c
62 0 -10 11 -12 13 u=8 fill=1 (0.8128 0.8128 0)
63 3 -6.55 60 -61 62 -63 #8 u=8 $ Zr flow channel
64 4 -1. 129 -65 66 -118 #8 #9 u=8 $ water
65 5 -7.84 118:-129:65:-66 u=8 $ Steel
c
c Type F box - No Boral Panels or fuel
c
66 4 -1. 129 -65 66 -118 u=9 $ water
67 5 -7.84 118:-129:65:-66 u=9 $ Steel
c
68 4 -1.0 -34 35 -36 37 u=5 lat=1 fill=-7:6 -7:6 0:0
5 5 5 5 5 5 5 5 5 5 5 5 5 5
5 9 9 9 9 9 9 9 9 9 9 9 9 5
5 9 9 9 9 9 7 4 9 9 9 9 9 5
5 9 9 9 7 4 4 4 4 4 9 9 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 7 4 4 4 4 4 4 4 4 9 5
5 9 8 4 4 4 4 4 4 4 4 6 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 9 8 4 4 4 4 4 4 4 6 9 9 5
5 9 9 9 8 4 4 4 6 6 9 9 9 5
5 9 9 9 9 8 6 9 9 9 9 9 5
5 9 9 9 9 9 9 9 9 9 9 9 5
5 5 5 5 5 5 5 5 5 5 5 5 5
69 0 -41 50 -49 fill=5 (8.1661 8.1661 0)
c
274 4 -1.0 -41 360 -50 $ Water below Fuel (7.3 in.)
275 5 -7.84 -42 362 -360 $ MPC Steel below Fuel (2.5 in.)
276 5 -7.84 -205 300 -362 $ Transfer Cask Steel (2.0 in.)
277 8 -11.34 -205 301 -300 $ Transfer Cask Lead (2.5 in.)
278 5 -7.84 -205 302 -301 $ Transfer Cask Steel (1.0 in.)
c
279 4 -1.0 -41 49 -361 $ Water above Fuel (8.46 in.)

```

280	5	-7.84	-42	361	-363	\$ MPC Steel above Fuel (10.0 in)
281	4	-1.0	-42	363	-400	\$ Water (1.5 in.)
282	5	-7.84	-205	400	-401	\$ Transfer Cask Steel (0.75 in.)
283	9	-1.61	-205	401	-402	\$ Transfer Cask Neutron Shield (3.25 in.)
284	5	-7.84	-205	402	-403	\$ Transfer Cask Steel (0.5 in.)
c						
290	5	-7.84	41 -42	360 -361		\$ Radial Steel - MPC shell
291	4	-1.00	42 -200	362 -400		\$ Radial Water
292	5	-7.84	200 -201	362 -400		\$ Radial Steel - inner shell of Trnsfr Cask
293	8	-11.34	201 -202	362 -400		\$ Radial Lead - Transfer Cask lead
294	5	-7.84	202 -203	362 -400		\$ Radial Steel - outer shell of Trnsfr Cask
295	4	-1.00	203 -204	362 -400		\$ Radial Water - Water Jacket
296	5	-7.84	204 -205	362 -400		\$ Radial Steel - outer shell of Water Jacket
c						
500	4	-1.00	500 -501 -505	(205 :-302: 403)		\$ outer water reflector
501	0		505 :-500: 501			\$ outside world
1	cz		0.5283			\$ Fuel OD
2	cz		0.5398			\$ Clad ID
3	cz		0.6134			\$ Clad OD
4	cz		0.6744			\$ Thimble ID
5	cz		0.7506			\$ Thimble OD
6	px		0.8128			\$ Pin Pitch
7	px		-0.8128			
8	py		0.8128			
9	py		-0.8128			
10	px		6.6231			\$ Channel ID
11	px		-6.6231			
12	py		6.6231			
13	py		-6.6231			
14	py		7.8016			
15	py		7.8155			
16	py		7.8410			
17	py		8.0467			
18	py		8.0721			
118	py		8.0861			
20	px		-6.0325			
21	px		-6.2230			
23	px		6.0325			
24	px		6.2230			
25	px		-7.8016			
26	px		-7.8155			
27	px		-7.8410			
28	px		-8.0467			
29	px		-8.0721			
129	px		-8.0861			
30	py		-6.0325			
31	py		6.0325			
32	py		-6.2230			
33	py		6.2230			
34	px		7.6111			
35	px		-8.7211			
36	py		8.7211			
37	py		-7.6111			
49	pz		381.			\$ Top of Active Fuel
50	pz		0			\$ Start of Active Fuel
60	px		-6.9279			\$ Channel OD
61	px		6.9279			
62	py		-6.9279			
63	py		6.9279			
64	px		-7.6111			\$ Cell Box ID
65	px		7.6111			

```

66  py      -7.6111
67  py       7.6111
360 pz     -18.54  $ lower water thkness = 7.30 in.
361 pz     402.49  $ upper water thkness = 8.46 in.
362 pz     -24.892 $ thkness of MPC baseplate = 2.5 in.
363 pz     427.89  $ thkness of MPC lid = 10. in.
41  cz      85.57  $ I.D. = 67.375 in.
42  cz      86.84  $ I.D. = 68.375 in.
200 cz      87.31  $ I.D. = 68.75 in.
201 cz      89.22  $ I.D. = 70.25 in.
202 cz     100.65  $ I.D. = 79.25 in.
203 cz     103.19  $ I.D. = 81.25 in.
204 cz     116.80  $ I.D. = 91.97 in.
205 cz     118.07  $ I.D. = 92.972 in.
300 pz     -29.97  $ thkness steel - 2.0 in.
301 pz     -36.32  $ thkness lead - 2.5 in.
302 pz     -38.86  $ thkness steel - 1.0 in.
400 pz     431.70  $ thkness water - 1.5 in.
401 pz     433.61  $ thkness steel - 0.75 in.
402 pz     441.87  $ thkness neutron shield - 3.25 in.
403 pz     443.14  $ thkness steel - 0.5 in.
c
*500 pz  -68.86  $ lower boundary
*501 pz  473.14  $ upper boundary
*505 cz  148.07  $ outer radial boundary

imp:n      1 87r 0
kcode     10000 0.94 20 120
c
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3 h 0 381.
sp3 0 1
c
c
si4 s
      15 16
      13 14 15 16 17 18
      12 13 14 15 16 17 18 19
      12 13 14 15 16 17 18 19
      11 12 13 14 15 16 17 18 19 20
      11 12 13 14 15 16 17 18 19 20
      12 13 14 15 16 17 18 19
      12 13 14 15 16 17 18 19
      13 14 15 16 17 18
      15 16

sp4 1 67r
c
ds5 s
      30 30
      29 29 29 29 29 29
      28 28 28 28 28 28 28 28
      27 27 27 27 27 27 27 27
      26 26 26 26 26 26 26 26 26
      25 25 25 25 25 25 25 25 25
      24 24 24 24 24 24 24 24
      23 23 23 23 23 23 23 23
      22 22 22 22 22 22
      21 21

c
si11 -80.6831 -67.6783
si12 -64.1985 -51.1937

```

si13 -47.7139 -34.7091  
 si14 -31.2293 -18.2245  
 si15 -14.7447 -1.7399  
 si16 1.7399 14.7447  
 si17 18.2245 31.2293  
 si18 34.7091 47.7139  
 si19 51.1937 64.1985  
 si20 67.6783 80.6831

c  
 si21 -80.6831 -67.6783  
 si22 -64.1985 -51.1937  
 si23 -47.7139 -34.7091  
 si24 -31.2293 -18.2245  
 si25 -14.7447 -1.7399  
 si26 1.7399 14.7447  
 si27 18.2245 31.2293  
 si28 34.7091 47.7139  
 si29 51.1937 64.1985  
 si30 67.6783 80.6831

sp11 0 1  
 sp12 0 1  
 sp13 0 1  
 sp14 0 1  
 sp15 0 1  
 sp16 0 1  
 sp17 0 1  
 sp18 0 1  
 sp19 0 1  
 sp20 0 1  
 sp21 0 1  
 sp22 0 1  
 sp23 0 1  
 sp24 0 1  
 sp25 0 1  
 sp26 0 1  
 sp27 0 1  
 sp28 0 1  
 sp29 0 1  
 sp30 0 1

c  
 m1 92235.50c -0.03702 \$ 4.20% E Fuel  
 92238.50c -0.84448  
 8016.50c -0.1185  
 m3 40000.56c 1. \$ Zr Clad  
 m4 1001.50c 0.6667 \$ Water  
 8016.50c 0.3333  
 m5 24000.50c 0.01761 \$ Steel  
 25055.50c 0.001761  
 26000.55c 0.05977  
 28000.50c 0.008239  
 m6 5010.50c 8.0707E-03 \$ Boral  
 5011.50c 3.2553E-02  
 6000.50c 1.0146E-02  
 13027.50c 3.8054E-02  
 m7 13027.50c 1. \$ Al Clad  
 m8 82000.50c 1.0 \$ Lead  
 m9 6000.50c -27.660 \$ Neutron Shield Holtite-A (NS-4-FR)  
 1001.50c -5.920  
 13027.50c -21.285  
 7014.50c -1.98  
 8016.50c -42.372  
 5010.50c -0.141

```

                    5011.50c   -0.642
mt4                lwtr.01t
prdmj              j   -120   j    2
fm4               1000   1    -6
f4:n              1
sd4               1000
e4  1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
     1.500E-09   2.000E-09   2.500E-09   3.000E-09
     4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
     3.000E-08   4.000E-08   5.000E-08   6.000E-08   7.000E-08
     8.000E-08   9.000E-08   1.000E-07   1.250E-07   1.500E-07
     1.750E-07   2.000E-07   2.250E-07   2.500E-07   2.750E-07
     3.000E-07   3.250E-07   3.500E-07   3.750E-07   4.000E-07
     4.500E-07   5.000E-07   5.500E-07   6.000E-07   6.250E-07
     6.500E-07   7.000E-07   7.500E-07   8.000E-07   8.500E-07
     9.000E-07   9.250E-07   9.500E-07   9.750E-07   1.000E-06
     1.010E-06   1.020E-06   1.030E-06   1.040E-06   1.050E-06
     1.060E-06   1.070E-06   1.080E-06   1.090E-06   1.100E-06
     1.110E-06   1.120E-06   1.130E-06   1.140E-06   1.150E-06
     1.175E-06   1.200E-06   1.225E-06   1.250E-06   1.300E-06
     1.350E-06   1.400E-06   1.450E-06   1.500E-06   1.590E-06
     1.680E-06   1.770E-06   1.860E-06   1.940E-06   2.000E-06
     2.120E-06   2.210E-06   2.300E-06   2.380E-06   2.470E-06
     2.570E-06   2.670E-06   2.770E-06   2.870E-06   2.970E-06
     3.000E-06   3.050E-06   3.150E-06   3.500E-06   3.730E-06
     4.000E-06   4.750E-06   5.000E-06   5.400E-06   6.000E-06
     6.250E-06   6.500E-06   6.750E-06   7.000E-06   7.150E-06
     8.100E-06   9.100E-06   1.000E-05   1.150E-05   1.190E-05
     1.290E-05   1.375E-05   1.440E-05   1.510E-05   1.600E-05
     1.700E-05   1.850E-05   1.900E-05   2.000E-05   2.100E-05
     2.250E-05   2.500E-05   2.750E-05   3.000E-05   3.125E-05
     3.175E-05   3.325E-05   3.375E-05   3.460E-05   3.550E-05
     3.700E-05   3.800E-05   3.910E-05   3.960E-05   4.100E-05
     4.240E-05   4.400E-05   4.520E-05   4.700E-05   4.830E-05
     4.920E-05   5.060E-05   5.200E-05   5.340E-05   5.900E-05
     6.100E-05   6.500E-05   6.750E-05   7.200E-05   7.600E-05
     8.000E-05   8.200E-05   9.000E-05   1.000E-04   1.080E-04
     1.150E-04   1.190E-04   1.220E-04   1.860E-04   1.925E-04
     2.075E-04   2.100E-04   2.400E-04   2.850E-04   3.050E-04
     5.500E-04   6.700E-04   6.830E-04   9.500E-04   1.150E-03
     1.500E-03   1.550E-03   1.800E-03   2.200E-03   2.290E-03
     2.580E-03   3.000E-03   3.740E-03   3.900E-03   6.000E-03
     8.030E-03   9.500E-03   1.300E-02   1.700E-02   2.500E-02
     3.000E-02   4.500E-02   5.000E-02   5.200E-02   6.000E-02
     7.300E-02   7.500E-02   8.200E-02   8.500E-02   1.000E-01
     1.283E-01   1.500E-01   2.000E-01   2.700E-01   3.300E-01
     4.000E-01   4.200E-01   4.400E-01   4.700E-01   4.995E-01
     5.500E-01   5.730E-01   6.000E-01   6.700E-01   6.790E-01
     7.500E-01   8.200E-01   8.611E-01   8.750E-01   9.000E-01
     9.200E-01   1.010E+00   1.100E+00   1.200E+00   1.250E+00
     1.317E+00   1.356E+00   1.400E+00   1.500E+00   1.850E+00
     2.354E+00   2.479E+00   3.000E+00   4.304E+00   4.800E+00
     6.434E+00   8.187E+00   1.000E+01   1.284E+01   1.384E+01
     1.455E+01   1.568E+01   1.733E+01   2.000E+01

```

HI-STORM Storage Cask containing MPC24, 17x17 assembly @ 4.0 wt% Enrich.

```
c
c
c
c MPC-24/24E cell configuration
c
c HI-STORM with active length 150 inch
c
c
c Cask Input Preprocessor
c cskinp 17a 17a mpc24n mpc24n historm historm 4.0 4sf7a45 empty
c ----- cpp\17a.bat
c   added 17a.ce
c   added 17a.su
c   added 17a.sp
c ----- cpp\mpc24n.bat
c   added mpc24n.co
c   added mpc24n.ce
c   added mpc24n.su
c   added mpc24n.sp
c ----- cpp\historm.bat
c   added historm.co
c   added historm.ce
c   added historm.su
c   added historm.sp
c end of comments
c
c start of cells
c
c 17x17a
c
c number of cells: 6
c cell numbers:      1 to 7
c univers numbers:  1 to 3
c surface numbers:  1 to 9
c
c number of cells: 1
1   1 -10.522   -1  u=2      $ fuel
2   4 -1.0      1  -2  u=2      $ gap
3   3 -6.55     2  -3  u=2      $ Zr Clad
4   2 -0.0002   3  u=2      $ water in fuel region
5   2 -0.0002  -4:5    u=3      $ water in guide tubes
6   3 -6.55     4  -5    u=3      $ guide tubes
7   2 -0.0002  -6   +7   -8   +9   u=1  lat=1
fill= -9:9    -9:9    0:0
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 3 2 2 2 2 2 1
1 2 2 2 3 2 2 2 2 2 2 2 2 2 2 3 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 3 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 2 2 2 3 2 2 2 2 2 2 2 2 2 2 3 2 2 2 2 1
1 2 2 2 2 2 3 2 2 3 2 2 3 2 2 3 2 2 2 2 1
1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
```



```

1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
c
c   MPC-24
c
c number of cells: 102
c cell numbers :      400   to   699
c universe numbers :    4   to    9
c surface numbers :   400   to   699
c
c   Right Side
c
408    0          -410  411  -412  413      u=4 fill=1 (1)
409    5  -7.84    410  -424  413  -426      u=4
410    2 -0.0002    424  -428  448  -445      u=4
411    7  -2.7     428  -528  448  -445      u=4
412    6  -2.66   528  -532  448  -445      u=4
413    7  -2.7     532  -432  448  -445      u=4
414    2 -0.0002    432  -436  448  -445      u=4
415    5  -7.84    436  -440  448  -445      u=4
416    2 -0.0002    440          413      u=4
417    2 -0.0002    424  -440  413  -447      u=4
418    2 -0.0002    424  -440  446      u=4
419    5  -7.84    424  -440  447  -448      u=4
420    5  -7.84    424  -440  445  -446      u=4
c
c   Left Side
c
421    5  -7.84    425  -411  413          u=4
422    2 -0.0002    429  -425  448  -445      u=4
423    7  -2.7     529  -429  448  -445      u=4
424    6  -2.66   533  -529  448  -445      u=4
425    7  -2.7     433  -533  448  -445      u=4
426    2 -0.0002    437  -433  448  -445      u=4
427    5  -7.84    441  -437  448  -445      u=4
428    2 -0.0002          -441  413      u=4
429    2 -0.0002    441  -425  413  -447      u=4
430    2 -0.0002    441  -425  446      u=4
431    5  -7.84    441  -425  447  -448      u=4
432    5  -7.84    441  -425  445  -446      u=4
c
c   Top
c
433    5  -7.84    411  -410  412  -426      u=4
434    2 -0.0002    451  -452  426  -430      u=4
435    7  -2.7     451  -452  430  -530      u=4
436    6  -2.66    451  -452  530  -534      u=4
437    7  -2.7     451  -452  534  -434      u=4
438    2 -0.0002    451  -452  434  -438      u=4
439    5  -7.84    451  -452  438  -442      u=4
440    2 -0.0002    411  -424  442          u=4
441    2 -0.0002    411  -450  426  -442      u=4
442    2 -0.0002    453  -424  426  -442      u=4
443    5  -7.84    450  -451  426  -442      u=4
444    5  -7.84    452  -453  426  -442      u=4
c
c   Bottom
c
445    5  -7.84    427          -413      u=4
446    2 -0.0002    451  -452  431  -427      u=4
447    7  -2.7     451  -452  531  -431      u=4
448    6  -2.66    451  -452  535  -531      u=4

```

```

449 7 -2.7 451 -452 435 -535 u=4
450 2 -0.0002 451 -452 439 -435 u=4
451 5 -7.84 451 -452 443 -439 u=4
452 2 -0.0002 411 -443 u=4
453 2 -0.0002 411 -450 443 -427 u=4
454 2 -0.0002 453 443 -427 u=4
455 5 -7.84 450 -451 443 -427 u=4
456 5 -7.84 452 -453 443 -427 u=4
457 5 -7.84 425 -411 -427 u=4
458 2 -0.0002 -425 -427 u=4
c
c TYPE B CELL - Short Boral on top and right
c
c Right Side
c
459 0 -410 411 -412 413 u=5 fill=1 (1)
460 5 -7.84 410 -424 413 -426 u=5
470 2 -0.0002 424 -428 548 -545 u=5
471 7 -2.7 428 -528 548 -545 u=5
472 6 -2.66 528 -532 548 -545 u=5
473 7 -2.7 532 -432 548 -545 u=5
474 2 -0.0002 432 -436 548 -545 u=5
475 5 -7.84 436 -440 548 -545 u=5
476 2 -0.0002 440 413 u=5
477 2 -0.0002 424 -440 413 -547 u=5
478 2 -0.0002 424 -440 546 u=5
479 5 -7.84 424 -440 547 -548 u=5
480 5 -7.84 424 -440 545 -546 u=5
c
c Left Side
c
481 5 -7.84 425 -411 413 u=5
482 2 -0.0002 429 -425 448 -445 u=5
483 7 -2.7 529 -429 448 -445 u=5
484 6 -2.66 533 -529 448 -445 u=5
485 7 -2.7 433 -533 448 -445 u=5
486 2 -0.0002 437 -433 448 -445 u=5
487 5 -7.84 441 -437 448 -445 u=5
488 2 -0.0002 -441 413 u=5
489 2 -0.0002 441 -425 413 -447 u=5
490 2 -0.0002 441 -425 446 u=5
491 5 -7.84 441 -425 447 -448 u=5
492 5 -7.84 441 -425 445 -446 u=5
c
c Top
c
493 5 -7.84 411 -410 412 -426 u=5
494 2 -0.0002 551 -552 426 -430 u=5
495 7 -2.7 551 -552 430 -530 u=5
496 6 -2.66 551 -552 530 -534 u=5
497 7 -2.7 551 -552 534 -434 u=5
498 2 -0.0002 551 -552 434 -438 u=5
499 5 -7.84 551 -552 438 -442 u=5
500 2 -0.0002 411 -424 442 u=5
501 2 -0.0002 411 -550 426 -442 u=5
502 2 -0.0002 553 -424 426 -442 u=5
503 5 -7.84 550 -551 426 -442 u=5
504 5 -7.84 552 -553 426 -442 u=5
c
c Bottom
c
505 5 -7.84 427 -413 u=5

```

```

506  2 -0.0002      451 -452  431 -427      u=5
507  7 -2.7        451 -452  531 -431      u=5
508  6 -2.66       451 -452  535 -531      u=5
509  7 -2.7        451 -452  435 -535      u=5
510  2 -0.0002      451 -452  439 -435      u=5
511  5 -7.84       451 -452  443 -439      u=5
512  2 -0.0002      411          -443      u=5
513  2 -0.0002      411 -450  443 -427      u=5
514  2 -0.0002      453          443 -427      u=5
515  5 -7.84       450 -451  443 -427      u=5
516  5 -7.84       452 -453  443 -427      u=5
517  5 -7.84       425 -411          -427      u=5
518  2 -0.0002          -425          -427      u=5
c
c
c
c   TYPE D CELL - Short Boral on left and bottom, different cell ID
c
c number of cells: 51
c
c   Right Side
c
1570  0          -1410  1411 -1412  1413      u=17 fill=1 (1)
1571  5 -7.84      1410 -1424  1413 -1426      u=17
1572  2 -0.0002      1424 -1428  1448 -1445      u=17
1573  7 -2.7        1428 -1528  1448 -1445      u=17
1574  6 -2.66       1528 -1532  1448 -1445      u=17
1575  7 -2.7        1532 -1432  1448 -1445      u=17
1576  2 -0.0002      1432 -1436  1448 -1445      u=17
1577  5 -7.84      1436 -1440  1448 -1445      u=17
1578  2 -0.0002      1440          1413      u=17
1579  2 -0.0002      1424 -1440  1413 -1447      u=17
1580  2 -0.0002      1424 -1440  1446          u=17
1581  5 -7.84      1424 -1440  1447 -1448      u=17
1582  5 -7.84      1424 -1440  1445 -1446      u=17
c
c   Left Side
c
1583  5 -7.84      1425 -1411  1413          u=17
1584  2 -0.0002      1429 -1425  1548 -1545      u=17
1585  7 -2.7        1529 -1429  1548 -1545      u=17
1586  6 -2.66       1533 -1529  1548 -1545      u=17
1587  7 -2.7        1433 -1533  1548 -1545      u=17
1588  2 -0.0002      1437 -1433  1548 -1545      u=17
1589  5 -7.84      1441 -1437  1548 -1545      u=17
1590  2 -0.0002          -1441  1413          u=17
1591  2 -0.0002      1441 -1425  1413 -1547      u=17
1592  2 -0.0002      1441 -1425  1546          u=17
1593  5 -7.84      1441 -1425  1547 -1548      u=17
1594  5 -7.84      1441 -1425  1545 -1546      u=17
c
c   Top
c
1595  5 -7.84      1411 -1410  1412 -1426      u=17
1596  2 -0.0002      1451 -1452  1426 -1430      u=17
1597  7 -2.7        1451 -1452  1430 -1530      u=17
1598  6 -2.66       1451 -1452  1530 -1534      u=17
1599  7 -2.7        1451 -1452  1534 -1434      u=17
1600  2 -0.0002      1451 -1452  1434 -1438      u=17
1601  5 -7.84      1451 -1452  1438 -1442      u=17
1602  2 -0.0002      1411 -1424  1442          u=17
1603  2 -0.0002      1411 -1450  1426 -1442      u=17

```

```

1604 2 -0.0002 1453 -1424 1426 -1442 u=17
1605 5 -7.84 1450 -1451 1426 -1442 u=17
1606 5 -7.84 1452 -1453 1426 -1442 u=17

```

```

c
c Bottom
c

```

```

1607 5 -7.84 1427 -1413 u=17
1608 2 -0.0002 1551 -1552 1431 -1427 u=17
1609 7 -2.7 1551 -1552 1531 -1431 u=17
1610 6 -2.66 1551 -1552 1535 -1531 u=17
1611 7 -2.7 1551 -1552 1435 -1535 u=17
1612 2 -0.0002 1551 -1552 1439 -1435 u=17
1613 5 -7.84 1551 -1552 1443 -1439 u=17
1614 2 -0.0002 1411 -1443 u=17
1615 2 -0.0002 1411 -1550 1443 -1427 u=17
1616 2 -0.0002 1553 1443 -1427 u=17
1617 5 -7.84 1550 -1551 1443 -1427 u=17
1618 5 -7.84 1552 -1553 1443 -1427 u=17
1619 5 -7.84 1425 -1411 -1427 u=17
1620 2 -0.0002 -1425 -1427 u=17

```

```

c
c number of cells: 29
c

```

```

c empty cell no boral, no top
c

```

```

751 2 -0.0002 -410 411 -412 413 u=14
752 5 -7.84 410 -424 413 -426 u=14
753 5 -7.84 425 -411 413 u=14
754 2 -0.0002 411 -410 412 -426 u=14
755 5 -7.84 427 -413 u=14
756 5 -7.84 425 -411 -427 u=14
757 2 -0.0002 411 426 u=14
758 2 -0.0002 411 -427 u=14
759 2 -0.0002 -425 413 u=14
760 2 -0.0002 424 413 -426 u=14
761 2 -0.0002 -425 -427 u=14

```

```

c
c
701 5 -7.84 701 -702 711 -713 u=9 $ steel post
702 5 -7.84 702 -703 711 -712 u=9 $ steel post

```

```

c
711 0 701 -705 711 -715 (702:713) (703:712)
fill=4 (13.8506 13.8506 0) u=9
712 0 704 (-706:-716) (705:715) -717 -710
fill=4 (17.9489 41.5518 0 0 1 0 -1 0 0 0 0 1) u=9
713 0 (705:715) -707 714 (-706:-716) 710
fill=4 (41.5518 17.9489 0 0 -1 0 1 0 0 0 0 1) u=9
714 0 701 -705 717 -719
fill=5 (13.8506 69.253 0) u=9
715 0 707 -709 711 -715
fill=5 (69.253 13.8506 0) u=9
716 0 706 -708 716 -718
fill=17 (45.6501 45.6501 0 -1 0 0 0 -1 0 0 0 1) u=9
717 0 705 -706 717 -719
fill=14 (41.5518 69.253 0) u=9
718 0 707 -709 715 -716
fill=14 (69.253 41.5518 0 0 1 0 1 0 0 0 0 1) u=9
719 0 701 -704 715 -717
fill=14 (-9.75233 41.5518 0 -1 0 0 0 1 0 0 0 1) u=9
720 0 705 -707 711 -714
fill=14 (41.5518 -9.75233 0 0 -1 0 1 0 0 0 0 1) u=9

```

```

721  2 -0.0002      (706:719) (708:718) (709:716) u=9
c
c
c
731  2 -0.0002      720  721   fill=9 (0 0 0) u=19
732  2 -0.0002     -720  721   fill=9 (0 0 0
      -1 0 0 0 1 0 0 0 1) u=19
733  2 -0.0002      720 -721   fill=9 (0 0 0
      1 0 0 0 -1 0 0 0 1) u=19
734  2 -0.0002     -720 -721   fill=9 (0 0 0
      -1 0 0 0 -1 0 0 0 1) u=19
c
673  0              -41              39 -40   fill=19
c
c number of cells: 19
374  2 -0.0002 -41      330 -39              $ Void below Fuel (4 in.)
375  5 -7.84  -309     332 -330             $ MPC Steel below Fuel (2.5 in.)
376  5 -7.84  -304     310 -332             $ Cask Steel (5.0 in.)
377  8 -2.35  -304     311 -310             $ Cask Concrete (17.0 in.)
378  5 -7.84  -304     312 -311             $ Cask Steel (2.0 in.)
c
379  2 -0.0002 -41      40 -331             $ Void above Fuel (6 in.)
380  5 -7.84  -309     331 -333             $ MPC Steel above Fuel (9.5 - 0.06 in)
381  4 -1.0    -309     333 -320             $ Water (1.0 in.)
382  5 -7.84  -304     320 -321             $ Cask Steel (1.25 in.)
383  8 -2.35  -304     321 -322             $ Cask Concrete (10.5 in.)
384  5 -7.84  -304     322 -323             $ Cask Steel (4.0 in.)
c
390  5 -7.84      41 -309  330 -331             $ Radial Steel - MPC shell
391  4 -1.00     309 -300  332 -320             $ Radial Water
392  5 -7.84     300 -301  332 -320             $ Radial Steel - overpack inner shell
394  5 -7.84     301 -302  332 -320             $ Radial Steel -
395  8 -2.35     302 -303  332 -320             $ Radial Shield - Concrete Overpack
396  5 -7.84     303 -304  332 -320             $ Radial Steel - overpack outer shell
c
300  4 -1.00     340 -341 -345             (304 :-312: 323)  $ outer water reflector
301  0              345 :-340: 341             $ outside world
c end of cells
c --- empty line

c --- empty line
c start of surfaces
1   cz          0.3922   $ fuel
2   cz          0.4001   $ clad ID
3   cz          0.4572   $ clad OD
4   cz          0.5613   $ guide ID
5   cz          0.6020   $ guide OD
6   px          0.6299   $ pin pitch
7   px         -0.6299
8   py          0.6299
9   py         -0.6299
c
c
c cell-id      8.98
c cell-pitch   10.906
c wall-thkns   5/16
c angle-thkns  5/16
c boral-gap    0.0035
c boral-gap-o  0.0035
c boral-thkns  0.075
c boral-clad   0.01
c sheathing    0.0235

```

```

c boral-wide      7.5
c boral-narrow   6.25
c
c gap size       1.09
c basket-od      67.335
c
410 px          11.40460 $x 8.98/2
411 px          -11.40460 $x {410} *-1
412 py          11.40460 $x {410}
413 py          -11.40460 $x {411}
416 px          13.85062 $x (10.906 + 5/16 - 5/16) /2
417 px          -13.85062 $x -10.906 + {416}
418 py          13.85062 $x {416}
419 py          -13.85062 $x {417}
424 px          12.19835 $x {410} + 5/16      $ angle
425 px          -12.19835 $x {411} - 5/16      $ box wall
426 py          12.19835 $x {412} + 5/16
427 py          -12.19835 $x {413} - 5/16
428 px          12.20724 $x {424} + 0.0035     $ wall to boral gap
429 px          -12.20724 $x {425} - 0.0035
430 py          12.20724 $x {426} + 0.0035
431 py          -12.20724 $x {427} - 0.0035
432 px          12.39774 $x {428} + 0.075      $ boral
433 px          -12.39774 $x {429} - 0.075
434 py          12.39774 $x {430} + 0.075
435 py          -12.39774 $x {431} - 0.075
436 px          12.40663 $x {432} + 0.0035     $ boral to sheathing gap
437 px          -12.40663 $x {433} - 0.0035
438 py          12.40663 $x {434} + 0.0035
439 py          -12.40663 $x {435} - 0.0035
440 px          12.46632 $x {436} + 0.0235     $ sheathing
441 px          -12.46632 $x {437} - 0.0235
442 py          12.46632 $x {438} + 0.0235
443 py          -12.46632 $x {439} - 0.0235
445 py          9.52500 $x 7.5/2
446 py          9.58469 $x {445} + 0.0235     $ sheathing
447 py          -9.58469 $x {446} *-1
448 py          -9.52500 $x {445} *-1
450 px          -9.58469 $x {447}
451 px          -9.52500 $x {448}
452 px          9.52500 $x {445}
453 px          9.58469 $x {446}
528 px          12.23264 $x {428} + 0.01      $ Aluminum on the outside of boral
529 px          -12.23264 $x {429} - 0.01
530 py          12.23264 $x {430} + 0.01
531 py          -12.23264 $x {431} - 0.01
532 px          12.37234 $x {432} - 0.01
533 px          -12.37234 $x {433} + 0.01
534 py          12.37234 $x {434} - 0.01
535 py          -12.37234 $x {435} + 0.01
545 py          7.93750 $x 6.25/2
546 py          7.99719 $x {545} + 0.0235     $ sheathing
547 py          -7.99719 $x {546} *-1
548 py          -7.93750 $x {545} *-1
550 px          -7.99719 $x {547}
551 px          -7.93750 $x {548}
552 px          7.93750 $x {545}
553 px          7.99719 $x {546}
c
c cell-id-2     8.98
c gap-o         1.09
c

```

```

701 px -5.0
702 px 1.90627 $x (10.906 - 8.98)/2 - 5/16 + 0.1
703 px 3.45694 $x 2.722/2
704 px 4.09829 $x 10.906 - 8.98 - 5/16
705 px 27.70124 $x 10.906
706 px 31.79953 $x 2 * 10.906 - (8.98+8.98)/2 - 5/16
707 px 55.40248 $x 2 * 10.906
708 px 59.50077 $x {707} + {704}
709 px 83.10372 $x 3 * 10.906
710 p 1 -1 0 0.1 $ diagonal x=y, offset by 0.1 to avoid intersecting
corners
711 py -4.99999 $x {701}
712 py 1.90627 $x {702}
713 py 3.45694 $x {703}
714 py 4.09829 $x {704}
715 py 27.70124 $x {705}
716 py 31.79953 $x {706}
717 py 55.40248 $x {707}
718 py 59.50077 $x {708}
719 py 83.10372 $x {709}
720 px 0.0
721 py 0.0
1410 px 11.40460 $x 8.98/2
1411 px -11.40460 $x {1410} *-1
1412 py 11.40460 $x {1410}
1413 py -11.40460 $x {1411}
1424 px 12.19835 $x {1410} + 5/16 $ angle
1425 px -12.19835 $x {1411} - 5/16 $ box wall
1426 py 12.19835 $x {1412} + 5/16
1427 py -12.19835 $x {1413} - 5/16
1428 px 12.20724 $x {1424} + 0.0035 $ wall to boral gap
1429 px -12.20724 $x {1425} - 0.0035
1430 py 12.20724 $x {1426} + 0.0035
1431 py -12.20724 $x {1427} - 0.0035
1432 px 12.39774 $x {1428} + 0.075 $ boral
1433 px -12.39774 $x {1429} - 0.075
1434 py 12.39774 $x {1430} + 0.075
1435 py -12.39774 $x {1431} - 0.075
1436 px 12.40663 $x {1432} + 0.0035 $ boral to sheathing gap
1437 px -12.40663 $x {1433} - 0.0035
1438 py 12.40663 $x {1434} + 0.0035
1439 py -12.40663 $x {1435} - 0.0035
1440 px 12.46632 $x {1436} + 0.0235 $ sheathing
1441 px -12.46632 $x {1437} - 0.0235
1442 py 12.46632 $x {1438} + 0.0235
1443 py -12.46632 $x {1439} - 0.0235
1444 py 9.52500 $x 7.5/2
1445 py 9.58469 $x {1445} + 0.0235 $ sheathing
1446 py -9.58469 $x {1446} *-1
1447 py -9.52500 $x {1445} *-1
1448 py -9.58469 $x {1447}
1449 px -9.52500 $x {1448}
1450 px 9.52500 $x {1445}
1451 px 9.58469 $x {1446}
1452 px 12.23264 $x {1428} + 0.01 $ Aluminum on the outside of boral
1453 px -12.23264 $x {1429} - 0.01
1530 py 12.23264 $x {1430} + 0.01
1531 py -12.23264 $x {1431} - 0.01
1532 px 12.37234 $x {1432} - 0.01
1533 px -12.37234 $x {1433} + 0.01
1534 py 12.37234 $x {1434} - 0.01
1535 py -12.37234 $x {1435} + 0.01

```

```

1545 py 7.93750 $x 6.25/2
1546 py 7.99719 $x {1545} + 0.0235 $ sheathing
1547 py -7.99719 $x {1546} *-1
1548 py -7.93750 $x {1545} *-1
1550 px -7.99719 $x {1547}
1551 px -7.93750 $x {1548}
1552 px 7.93750 $x {1545}
1553 px 7.99719 $x {1546}
39 pz 0.
40 pz 381.0 $ 150 inch active fuel length
330 pz -10.16 $ lower water thkness = 4 in.
331 pz 396.24 $ upper water thkness = 6 in.
332 pz -16.51 $ thkness of MPC baseplate = 2.5 in.
333 pz 420.02 $ thkness of MPC lid = 9.5 -0.06 in.
41 cz 85.57 $ I.D. = 67.37 in
309 cz 86.84 $ I.D. = 68.375 in.
300 cz 93.35 $ I.D. = 73.50 in.
301 cz 96.52 $ I.D. = 76.00 in.
302 cz 98.43 $ I.D. = 77.50 in.
303 cz 166.37 $ I.D. = 131.00 in.
304 cz 168.28 $ I.D. = 132.50 in.
310 pz -29.21 $ thkness steel - 5.0 in.
311 pz -72.39 $ thkness concrete - 17.0 in.
312 pz -77.47 $ thkness steel - 2.0 in.
320 pz 422.76 $ thkness water - 1.0 in.
321 pz 425.94 $ thkness steel - 1.25 in.
322 pz 452.61 $ thkness concrete - 10.5 in.
323 pz 462.765 $ thkness steel - 4.0 in.
c
*340 pz -107.47 $ lower boundary
*341 pz 492.765 $ upper boundary
*345 cz 198.28 $ outer radial boundary
c end of surfaces
c --- empty line

c --- empty line
tr1 0 0 0
kcode 10000 .94 20 120
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
sp3 0 1
c
si4 s 13 14
12 13 14 15
11 12 13 14 15 16
11 12 13 14 15 16
12 13 14 15
13 14

sp4 1 23r
c
ds5 s 26 26
25 25 25 25
24 24 24 24 24 24
23 23 23 23 23 23
22 22 22 22
21 21

c
si11 -79.25435 -57.61355
si12 -51.88077 -30.23997
si13 -24.50719 -2.86639

```



```

si14  2.86639  24.50719
si15  30.23997  51.88077
si16  57.61355  79.25435
c
si21  -79.25435 -57.61355
si22  -51.88077 -30.23997
si23  -24.50719 -2.86639
si24  2.86639  24.50719
si25  30.23997  51.88077
si26  57.61355  79.25435
c
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
c
m3      40000.56c  1.          $ Zr Clad
m4      1001.50c  0.6667         $ Water
        8016.50c  0.3333
m5      24000.50c  0.01761        $ Steel
        25055.50c  0.001761
        26000.55c  0.05977
        28000.50c  0.008239
m6      5010.50c  -0.054427      $ Boral Central Section @ 0.02 g/cmsq
        5011.50c  -0.241373
        13027.50c -0.6222
        6000.50c  -0.0821
m7      13027.50c  1.0
mt4     lwtr.01t
prdmp   j  -120  j  2
fm4     1000  1  -6
f4:n    1
sd4     1000
e4      1.000E-11  1.000E-10  5.000E-10  7.500E-10  1.000E-09  1.200E-09
        1.500E-09  2.000E-09  2.500E-09  3.000E-09
        4.700E-09  5.000E-09  7.500E-09  1.000E-08  2.530E-08
        3.000E-08  4.000E-08  5.000E-08  6.000E-08  7.000E-08
        8.000E-08  9.000E-08  1.000E-07  1.250E-07  1.500E-07
        1.750E-07  2.000E-07  2.250E-07  2.500E-07  2.750E-07
        3.000E-07  3.250E-07  3.500E-07  3.750E-07  4.000E-07
        4.500E-07  5.000E-07  5.500E-07  6.000E-07  6.250E-07
        6.500E-07  7.000E-07  7.500E-07  8.000E-07  8.500E-07
        9.000E-07  9.250E-07  9.500E-07  9.750E-07  1.000E-06
        1.010E-06  1.020E-06  1.030E-06  1.040E-06  1.050E-06
        1.060E-06  1.070E-06  1.080E-06  1.090E-06  1.100E-06
        1.110E-06  1.120E-06  1.130E-06  1.140E-06  1.150E-06
        1.175E-06  1.200E-06  1.225E-06  1.250E-06  1.300E-06
        1.350E-06  1.400E-06  1.450E-06  1.500E-06  1.590E-06
        1.680E-06  1.770E-06  1.860E-06  1.940E-06  2.000E-06
        2.120E-06  2.210E-06  2.300E-06  2.380E-06  2.470E-06
        2.570E-06  2.670E-06  2.770E-06  2.870E-06  2.970E-06
        3.000E-06  3.050E-06  3.150E-06  3.500E-06  3.730E-06
        4.000E-06  4.750E-06  5.000E-06  5.400E-06  6.000E-06
        6.250E-06  6.500E-06  6.750E-06  7.000E-06  7.150E-06

```

8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05
1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05
1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05
2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05
3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05
3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05
4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05
4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05
6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05
8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04
1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04
2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04
5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03
1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03
2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03
8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02
3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02
7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	
si3	h 0 381.00			
m2	8016.50c	-1.0		
m8	13027.50c	-0.048	\$ Concrete	
	14000.50c	-0.315		
	8016.50c	-0.500		
	1001.50c	-0.006		
	11023.50c	-0.017		
	20000.50c	-0.083		
	26000.55c	-0.012		
	19000.50c	-0.019		
mt8	lwtr.01t			
imp:n 1 206r 0				
c fuel enrichment 4.0 %				
m1	92235.50c	-0.03526		
	92238.50c	-0.84624		
	8016.50c	-0.11850		
c end of file				
c				

HI-STORM Storage Cask containing MPC68, 08x08 assembly @ 4.2 wt% Enrich.

c MPC68 reflected w/60cm of water, 0.0279 g/cmsq B-10 in Boral

c

c

1 1 -10.522 -1 u=2 \$ fuel  
2 2 -0.0002 1 -2 u=2 \$ gap  
3 2 -0.0002 2 -3 u=2 \$ Zr Clad  
4 2 -0.0002 3 u=2 \$ water in fuel region  
5 2 -0.0002 -4:5 u=3 \$ water in guide tubes  
6 2 -0.0002 4 -5 u=3 \$ guide tubes  
7 2 -0.0002 -6 +7 -8 +9 u=1 lat=1

fill= -5:4 -5:4 0:0

1 1 1 1 1 1 1 1 1  
1 2 2 2 2 2 2 2 2  
1 2 2 2 2 2 2 2 2  
1 2 2 2 2 2 2 2 2  
1 2 2 2 3 2 2 2 2  
1 2 2 2 2 3 2 2 2  
1 2 2 2 2 2 2 2 2  
1 2 2 2 2 2 2 2 2  
1 2 2 2 2 2 2 2 2  
1 1 1 1 1 1 1 1 1

c

C BOX TYPE R

c

8 0 -10 11 -12 13 u=4 fill=1 (0.8128 0.8128 0)  
9 3 -6.55 60 -61 62 -63 #8 u=4 \$ Zr flow channel  
10 2 -0.0002 64 -65 66 -67 #8 #9 u=4 \$ water  
11 5 -7.84 20 -23 67 -14 u=4 \$ 0.075" STEEL  
12 2 -0.0002 20 -23 14 -15 u=4 \$ WATER POCKET  
13 7 -2.7 20 -23 15 -16 u=4 \$ Al CLAD  
14 6 -2.66 20 -23 16 -17 u=4 \$ BORAL Absorber  
15 7 -2.7 20 -23 17 -18 u=4 \$ Al Clad  
16 2 -0.0002 20 -23 18 -118 u=4 \$ Water  
17 5 -7.84 118:-129:65:-66 u=4 \$ Steel  
18 2 -0.0002 64 -21 67 -118 u=4 \$ Water  
19 2 -0.0002 24 -65 67 -118 u=4 \$ water  
20 5 -7.84 21 -20 67 -118 u=4 \$ Steel  
21 5 -7.84 23 -24 67 -118 u=4 \$ Steel  
22 2 -0.0002 129 -64 33 -118 u=4 \$ Water

c

23 5 -7.84 25 -64 30 -31 u=4 \$ Steel  
24 2 -0.0002 26 -25 30 -31 u=4 \$ Water  
25 7 -2.7 27 -26 30 -31 u=4 \$ Al clad  
26 6 -2.66 28 -27 30 -31 u=4 \$ Boral  
27 7 -2.7 29 -28 30 -31 u=4 \$ Al clad  
28 2 -0.0002 129 -29 30 -31 u=4 \$ water  
29 5 -7.84 129 -64 32 -30 u=4 \$ Steel ends  
30 5 -7.84 129 -64 31 -33 u=4 \$ Steel ends  
31 2 -0.0002 129 -64 66 -32 u=4 \$ Water

c

c Type A box - Boral only on left side

c

32 0 -10 11 -12 13 u=6 fill=1 (0.8128 0.8128 0)  
33 3 -6.55 60 -61 62 -63 #8 u=6 \$ Zr flow channel  
34 2 -0.0002 64 -65 66 -118 #8 #9 u=6 \$ water  
35 5 -7.84 118:-129:65:-66 u=6 \$ Steel  
36 2 -0.0002 129 -64 67 -118 u=6 \$ Water  
37 5 -7.84 25 -64 30 -31 u=6 \$ Steel  
38 2 -0.0002 26 -25 30 -31 u=6 \$ Water  
39 7 -2.7 27 -26 30 -31 u=6 \$ Al clad

```

40 6 -2.66 28 -27 30 -31 u=6 $ Boral
41 7 -2.7 29 -28 30 -31 u=6 $ Al clad
42 2 -0.0002 129 -29 30 -31 u=6 $ water
43 2 -0.0002 129 -64 33 -67 u=6 $ Water
44 5 -7.84 129 -64 32 -30 u=6 $ Steel ends
45 5 -7.84 129 -64 31 -33 u=6 $ Steel ends
46 2 -0.0002 129 -64 66 -32 u=6 $ Water
c
c Type B box - Boral on Top only
c
47 0 -10 11 -12 13 u=7 fill=1 (0.8128 0.8128 0)
48 3 -6.55 60 -61 62 -63 #8 u=7 $ Zr flow channel
49 2 -0.0002 64 -65 66 -67 #8 #9 u=7 $ water
50 5 -7.84 20 -23 67 -14 u=7 $ 0.075" STEEL
51 2 -0.0002 20 -23 14 -15 u=7 $ WATER POCKET
52 7 -2.7 20 -23 15 -16 u=7 $ Al CLAD
53 6 -2.66 20 -23 16 -17 u=7 $ BORAL Absorber
54 7 -2.7 20 -23 17 -18 u=7 $ water
55 2 -0.0002 20 -23 18 -118 u=7 $ Water
56 5 -7.84 118:-129:65:-66 u=7 $ Steel
57 2 -0.0002 64 -21 67 -118 u=7 $ Water
58 2 -0.0002 24 -65 67 -118 u=7 $ water
59 5 -7.84 21 -20 67 -118 u=7 $ Steel
60 5 -7.84 23 -24 67 -118 u=7 $ Steel
61 2 -0.0002 129 -64 66 -118 u=7 $ Water
c
c Type E box - No Boral Panels
c
62 0 -10 11 -12 13 u=8 fill=1 (0.8128 0.8128 0)
63 3 -6.55 60 -61 62 -63 #8 u=8 $ Zr flow channel
64 2 -0.0002 129 -65 66 -118 #8 #9 u=8 $ water
65 5 -7.84 118:-129:65:-66 u=8 $ Steel
c
c Type F box - No Boral Panels or fuel
c
66 2 -0.0002 129 -65 66 -118 u=9 $ water
67 5 -7.84 118:-129:65:-66 u=9 $ Steel
c
68 2 -0.0002 -34 35 -36 37 u=5 lat=1 fill=-7:6 -7:6 0:0
5 5 5 5 5 5 5 5 5 5 5 5 5
5 9 9 9 9 9 9 9 9 9 9 9 5
5 9 9 9 9 9 7 4 9 9 9 9 5
5 9 9 9 7 4 4 4 4 4 9 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 7 4 4 4 4 4 4 4 4 9 5
5 9 8 4 4 4 4 4 4 4 4 6 9 5
5 9 9 7 4 4 4 4 4 4 4 9 9 5
5 9 9 8 4 4 4 4 4 4 4 6 9 9 5
5 9 9 9 8 4 4 4 6 6 9 9 9 5
5 9 9 9 9 9 8 6 9 9 9 9 5
5 9 9 9 9 9 9 9 9 9 9 9 5
5 5 5 5 5 5 5 5 5 5 5 5
69 0 -41 50 -49 fill=5 (8.1661 8.1661 0)
c
274 2 -0.0002 -41 360 -50 $ space below Fuel (7.3 in.)
275 5 -7.84 -42 362 -360 $ MPC Steel below Fuel (2.5 in.)
276 5 -7.84 -204 300 -362 $ Cask Steel (5.0 in.)
277 8 -2.35 -204 301 -300 $ Cask Concrete (17.0 in.)
278 5 -7.84 -204 302 -301 $ Cask Steel (2.0 in.)
c
279 2 -0.0002 -41 49 -361 $ space above Fuel (8.46 in.)

```

280	5	-7.84	-42	361	-363	\$ MPC Steel above Fuel (10.0 in)
281	4	-1.00	-42	363	-400	\$ Water (1.0 in.)
282	5	-7.84	-204	400	-401	\$ Cask Steel (1.25 in.)
283	8	-2.35	-204	401	-402	\$ Cask Concrete (10.5 in.)
284	5	-7.84	-204	402	-403	\$ Cask Steel (4.0 in.)
c						
290	5	-7.84	41 -42	360 -361		\$ Radial Steel - MPC shell
291	4	-1.00	42 -200	362 -400		\$ Radial Water
292	5	-7.84	200 -201	362 -400		\$ Radial Steel - overpack inner shell
293	5	-7.84	201 -202	362 -400		\$ Radial Steel -
294	8	-2.35	202 -203	362 -400		\$ Radial Shield - Concrete Overpack
295	5	-7.84	203 -204	362 -400		\$ Radial Steel - overpack outer shell
c						
500	4	-1.00	500 -501 -505	(204 :-302: 403)		\$ outer water reflector
501	0		505 :-500: 501			\$ outside world
1	cz	0.5283				\$ Fuel OD
2	cz	0.5398				\$ Clad ID
3	cz	0.6134				\$ Clad OD
4	cz	0.6744				\$ Thimble ID
5	cz	0.7506				\$ Thimble OD
6	px	0.8128				\$ Pin Pitch
7	px	-0.8128				
8	py	0.8128				
9	py	-0.8128				
10	px	6.6231				\$ Channel ID
11	px	-6.6231				
12	py	6.6231				
13	py	-6.6231				
14	py	7.8016				
15	py	7.8155				
16	py	7.8410				
17	py	8.0467				
18	py	8.0721				
118	py	8.0861				
20	px	-6.0325				
21	px	-6.2230				
23	px	6.0325				
24	px	6.2230				
25	px	-7.8016				
26	px	-7.8155				
27	px	-7.8410				
28	px	-8.0467				
29	px	-8.0721				
129	px	-8.0861				
30	py	-6.0325				
31	py	6.0325				
32	py	-6.2230				
33	py	6.2230				
34	px	7.6111				
35	px	-8.7211				
36	py	8.7211				
37	py	-7.6111				
49	pz	381.				\$ Top of Active Fuel
50	pz	0				\$ Start of Active Fuel
60	px	-6.9279				\$ Channel OD
61	px	6.9279				
62	py	-6.9279				
63	py	6.9279				
64	px	-7.6111				\$ Cell Box ID
65	px	7.6111				
66	py	-7.6111				

```

67   py           7.6111
360  pz          -18.54  $ lower thkness = 7.30 in.
361  pz          402.49  $ upper thkness = 8.46 in.
362  pz          -24.892 $ thkness of MPC baseplate = 2.5 in.
363  pz          427.89  $ thkness of MPC lid = 10. in.
41   cz           85.57  $ I.D. = 67.375 in.
42   cz           86.84  $ I.D. = 68.375 in.
200  cz           93.35  $ I.D. = 73.50 in.
201  cz           96.52  $ I.D. = 76.00 in.
202  cz           98.43  $ I.D. = 77.50 in.
203  cz          166.37  $ I.D. = 131.00 in.
204  cz          168.28  $ I.D. = 132.50 in.
300  pz          -37.59  $ thkness steel - 5.0 in.
301  pz          -80.77  $ thkness concrete - 17.0 in.
302  pz          -85.85  $ thkness steel - 2.0 in.
400  pz          430.43  $ thkness water - 1.0 in.
401  pz          433.605 $ thkness steel - 1.25 in.
402  pz          460.28  $ thkness concrete - 10.5 in.
403  pz          465.355 $ thkness steel - 4.0 in.
c
*500 pz -115.85 $ lower boundary
*501 pz 495.355 $ upper boundary
*505 cz 198.28 $ outer radial boundary

imp:n      1 86r 0
kcode     10000 0.94 20 120
c
sdef par=1 erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3 h 0 381.
sp3 0 1
c
c
si4 s
      15 16
      13 14 15 16 17 18
      12 13 14 15 16 17 18 19
      12 13 14 15 16 17 18 19
      11 12 13 14 15 16 17 18 19 20
      11 12 13 14 15 16 17 18 19 20
      12 13 14 15 16 17 18 19
      12 13 14 15 16 17 18 19
      13 14 15 16 17 18
      15 16

sp4 1 67r
c
ds5 s
      30 30
      29 29 29 29 29 29
      28 28 28 28 28 28 28 28
      27 27 27 27 27 27 27 27
      26 26 26 26 26 26 26 26 26
      25 25 25 25 25 25 25 25 25
      24 24 24 24 24 24 24 24
      23 23 23 23 23 23 23 23
      22 22 22 22 22 22
      21 21

c
si11 -80.6831 -67.6783
si12 -64.1985 -51.1937
si13 -47.7139 -34.7091
si14 -31.2293 -18.2245

```

si15	-14.7447	-1.7399	
si16	1.7399	14.7447	
si17	18.2245	31.2293	
si18	34.7091	47.7139	
si19	51.1937	64.1985	
si20	67.6783	80.6831	
c			
si21	-80.6831	-67.6783	
si22	-64.1985	-51.1937	
si23	-47.7139	-34.7091	
si24	-31.2293	-18.2245	
si25	-14.7447	-1.7399	
si26	1.7399	14.7447	
si27	18.2245	31.2293	
si28	34.7091	47.7139	
si29	51.1937	64.1985	
si30	67.6783	80.6831	
sp11	0	1	
sp12	0	1	
sp13	0	1	
sp14	0	1	
sp15	0	1	
sp16	0	1	
sp17	0	1	
sp18	0	1	
sp19	0	1	
sp20	0	1	
sp21	0	1	
sp22	0	1	
sp23	0	1	
sp24	0	1	
sp25	0	1	
sp26	0	1	
sp27	0	1	
sp28	0	1	
sp29	0	1	
sp30	0	1	
c			
m1	92235.50c	-0.03702	\$ 4.20% E Fuel
	92238.50c	-0.84448	
	8016.50c	-0.1185	
m2	8016.50c	1.	\$ Void
m3	40000.56c	1.	\$ Zr Clad
m4	1001.50c	0.6667	\$ Water
	8016.50c	0.3333	
m5	24000.50c	0.01761	\$ Steel
	25055.50c	0.001761	
	26000.55c	0.05977	
	28000.50c	0.008239	
m6	5010.50c	8.0707E-03	\$ Boral
	5011.50c	3.2553E-02	
	6000.50c	1.0146E-02	
	13027.50c	3.8054E-02	
m7	13027.50c	1.	\$ Al Clad
m8	13027.50c	-0.0048	\$ Concrete
	14000.50c	-0.315	
	8016.50c	-0.500	
	1001.50c	-0.006	
	11023.50c	-0.017	
	20000.50c	-0.083	
	26000.55c	-0.012	
	19000.50c	-0.019	

```

mt4          lwtr.01t
mt8          lwtr.01t
prdmp       j   -60   j   2
fm4         1000   1   -6
f4:n        1
sd4         1000
e4          1.000E-11  1.000E-10  5.000E-10  7.500E-10  1.000E-09  1.200E-09
            1.500E-09  2.000E-09  2.500E-09  3.000E-09
            4.700E-09  5.000E-09  7.500E-09  1.000E-08  2.530E-08
            3.000E-08  4.000E-08  5.000E-08  6.000E-08  7.000E-08
            8.000E-08  9.000E-08  1.000E-07  1.250E-07  1.500E-07
            1.750E-07  2.000E-07  2.250E-07  2.500E-07  2.750E-07
            3.000E-07  3.250E-07  3.500E-07  3.750E-07  4.000E-07
            4.500E-07  5.000E-07  5.500E-07  6.000E-07  6.250E-07
            6.500E-07  7.000E-07  7.500E-07  8.000E-07  8.500E-07
            9.000E-07  9.250E-07  9.500E-07  9.750E-07  1.000E-06
            1.010E-06  1.020E-06  1.030E-06  1.040E-06  1.050E-06
            1.060E-06  1.070E-06  1.080E-06  1.090E-06  1.100E-06
            1.110E-06  1.120E-06  1.130E-06  1.140E-06  1.150E-06
            1.175E-06  1.200E-06  1.225E-06  1.250E-06  1.300E-06
            1.350E-06  1.400E-06  1.450E-06  1.500E-06  1.590E-06
            1.680E-06  1.770E-06  1.860E-06  1.940E-06  2.000E-06
            2.120E-06  2.210E-06  2.300E-06  2.380E-06  2.470E-06
            2.570E-06  2.670E-06  2.770E-06  2.870E-06  2.970E-06
            3.000E-06  3.050E-06  3.150E-06  3.500E-06  3.730E-06
            4.000E-06  4.750E-06  5.000E-06  5.400E-06  6.000E-06
            6.250E-06  6.500E-06  6.750E-06  7.000E-06  7.150E-06
            8.100E-06  9.100E-06  1.000E-05  1.150E-05  1.190E-05
            1.290E-05  1.375E-05  1.440E-05  1.510E-05  1.600E-05
            1.700E-05  1.850E-05  1.900E-05  2.000E-05  2.100E-05
            2.250E-05  2.500E-05  2.750E-05  3.000E-05  3.125E-05
            3.175E-05  3.325E-05  3.375E-05  3.460E-05  3.550E-05
            3.700E-05  3.800E-05  3.910E-05  3.960E-05  4.100E-05
            4.240E-05  4.400E-05  4.520E-05  4.700E-05  4.830E-05
            4.920E-05  5.060E-05  5.200E-05  5.340E-05  5.900E-05
            6.100E-05  6.500E-05  6.750E-05  7.200E-05  7.600E-05
            8.000E-05  8.200E-05  9.000E-05  1.000E-04  1.080E-04
            1.150E-04  1.190E-04  1.220E-04  1.860E-04  1.925E-04
            2.075E-04  2.100E-04  2.400E-04  2.850E-04  3.050E-04
            5.500E-04  6.700E-04  6.830E-04  9.500E-04  1.150E-03
            1.500E-03  1.550E-03  1.800E-03  2.200E-03  2.290E-03
            2.580E-03  3.000E-03  3.740E-03  3.900E-03  6.000E-03
            8.030E-03  9.500E-03  1.300E-02  1.700E-02  2.500E-02
            3.000E-02  4.500E-02  5.000E-02  5.200E-02  6.000E-02
            7.300E-02  7.500E-02  8.200E-02  8.500E-02  1.000E-01
            1.283E-01  1.500E-01  2.000E-01  2.700E-01  3.300E-01
            4.000E-01  4.200E-01  4.400E-01  4.700E-01  4.995E-01
            5.500E-01  5.730E-01  6.000E-01  6.700E-01  6.790E-01
            7.500E-01  8.200E-01  8.611E-01  8.750E-01  9.000E-01
            9.200E-01  1.010E+00  1.100E+00  1.200E+00  1.250E+00
            1.317E+00  1.356E+00  1.400E+00  1.500E+00  1.850E+00
            2.354E+00  2.479E+00  3.000E+00  4.304E+00  4.800E+00
            6.434E+00  8.187E+00  1.000E+01  1.284E+01  1.384E+01
            1.455E+01  1.568E+01  1.733E+01  2.000E+01

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## CHAPTER 7<sup>†</sup>: CONFINEMENT

### 7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STORM 100 System is provided by the MPC. The design of the HI-STORM 100 confinement boundary assures that there are no credible design basis events that would result in a radiological release to the environment. The HI-STORM 100 Overpack and HI-TRAC Transfer Cask are designed to provide physical protection for an MPC during normal, off-normal, and postulated accident conditions to assure that the integrity of the MPC confinement boundary is maintained. The inert atmosphere in the MPC and the passive heat removal capabilities of the HI-STORM 100 also assure that the SNF assemblies remain protected from degradation, which might otherwise lead to gross cladding ruptures during dry storage.

A detailed description of the confinement structures, systems, and components important to safety is provided in Chapter 2. The structural adequacy of the MPC is demonstrated by the analyses documented in Chapter 3. The physical protection of the MPC provided by the Overpack and the HI-TRAC Transfer Cask is demonstrated by the structural analyses documented in Chapter 3 and for off-normal and postulated accident conditions in Chapter 11. The heat removal capabilities of the HI-STORM 100 System are demonstrated by the thermal analyses documented in Chapter 4.

This chapter describes the HI-STORM 100 confinement boundary design and describes how the design satisfies the confinement requirements of 10CFR72 [7.0.1]. It also provides an evaluation of *the MPC confinement boundary as it relates to the criteria contained in Interim Staff Guidance (ISG)-18 as justification for determining that leakage from the confinement boundary is not credible and, therefore, no confinement analysis is required.* ~~postulated radiological releases to the environment under normal, off-normal, and accident conditions of storage to ensure compliance with the limits established by the regulations.~~

This chapter is in compliance with NUREG-1536 except as noted in Table 1.0.3.

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

## 7.1 CONFINEMENT BOUNDARY

The primary confinement boundary against the release of radionuclides is the cladding of the individual fuel rods. The spent fuel rods are protected from degradation by maintaining an inert gas atmosphere (helium) inside the MPC and keeping the fuel cladding temperatures below the design basis values specified in Chapter 2.

The HI-STORM 100 confinement boundary consists of any one of the ~~seven~~ fully-welded MPC designs described in Chapter 1. Each MPC is identical from a confinement perspective so the following discussion applies to all MPCs. The confinement boundary of the MPC consists of:

- MPC shell
- bottom baseplate
- MPC lid (including the vent and drain port cover plates)
- MPC closure ring
- associated welds

The above items form a totally seal-welded vessel for the storage of design basis spent fuel assemblies.

The MPC requires no valves, gaskets or mechanical seals for confinement. Figure 7.1.1 shows an elevation cross-section of the MPC confinement boundary. All components of the confinement boundary are Important to Safety, Category A, as specified in Table 2.2.6. The MPC confinement boundary is designed and fabricated in accordance with the ASME Code, Section III, Subsection NB [7.1.1] to the maximum extent practicable. Chapter 2 provides design criteria for the confinement design. Section 2.2.4 provides applicable Code requirements. ~~Exceptions—NRC-approved alternatives~~ to specific Code requirements with complete justifications are presented in Table 2.2.15.

### 7.1.1 Confinement Vessel

The HI-STORM 100 *System* confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The MPC is designed, fabricated, *inspected*, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1] ~~including certain NRC-approved alternatives to the maximum extent practicable~~. The MPC shell and baseplate assembly and basket structure are delivered to the loading facility as one complete component. The MPC lid, vent and drain port cover plates, and closure ring are supplied separately and are installed following fuel loading. The MPC lid and closure ring are welded to the upper part of the MPC shell ~~at the after fuel loading site~~ to provide redundant sealing of the confinement boundary. The vent and drain port cover plates are welded to the MPC lid after the lid is welded to the MPC. The welds forming the confinement boundary are described in detail in Section 7.1.3.

The MPC lid is made intentionally thick to minimize radiation exposure to workers during MPC closure operations, and is welded to the MPC shell. The vent and drain port cover plates are welded to the MPC lid following completion of MPC draining, moisture removal, and helium backfill activities to close the MPC vent and drain openings. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter. The combination of the welded MPC lid and closure ring form the redundant closure of the MPC.

Table 7.1.1 provides a summary of the design ratings for normal, off-normal and accident conditions for the MPC confinement vessel. Tables 1.2.2, 2.2.1, and 2.2.3 provide additional design basis information.

~~The design basis leakage rate for the MPC confinement boundary is provided in Table 7.1.1. The MPC shell and baseplate are helium leakage tested during fabrication in accordance with the requirements defined in Chapter 9. Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by liquid penetrant method (root and final), volumetrically examined (or, if volumetric examination is not performed, multi-layer liquid penetrant examination must be performed), helium leakage tested, and hydrostatically pressure tested. If the MPC lid weld is acceptable, the vent and drain port cover plates are welded in place, and examined by the liquid penetrant method (root and final), and a leakage rate test is performed. Finally, the MPC closure ring is installed, welded and inspected by the liquid penetrant method (root, if multiple pass, and final). Chapters 8, 9, and 12 provide procedural guidance, acceptance criteria, and operating controls Technical Specifications, respectively, for performance and acceptance of liquid penetrant examinations, volumetric examination, hydrostatic and pressure testing, and leakage rate testing of the field welds on the MPC.~~

After moisture removal, the MPC cavity is backfilled with helium. The helium backfill provides an inert atmosphere within the MPC cavity that precludes oxidation and hydride attack of the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and reduces the maximum fuel cladding temperatures. MPC inerting, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

### 7.1.2 Confinement Penetrations

The MPC penetrations are designed to prevent the release of radionuclides under all normal, off-normal and accident conditions of storage. Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and for fuel cool-down and MPC flooding during unloading operations. No other confinement penetrations exist in the MPC. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain connectors allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The MPC vent and drain ports are sealed by cover plates which that are seal welded to the MPC lid. No credit is taken for the seal

provided by the vent and drain port *caps*. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, providing the redundant closure of the MPC vessel. The redundant closures of the MPC satisfy the requirements of 10CFR72.236(e) [7.0.1].

The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection.

### 7.1.3 Seals and Welds

The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], *with certain NRC-approved alternatives*. ~~to the maximum extent practicable~~. The MPC has no bolted closures or mechanical seals. Section 7.1.1 describes the design of the confinement vessel welds. The welds forming the confinement boundary are summarized in Table 7.1.2.

Confinement boundary welds are performed, inspected, and tested in accordance with the applicable requirements of ASME Section III, Subsection NB [7.1.1] *with certain NRC-approved alternatives*. ~~to the maximum extent practicable~~. The use of multi-pass welds, root pass, for multiple pass welds, and final surface liquid penetrant inspection, and volumetric examination essentially eliminates the chance of a pinhole leak through the weld. If volumetric examination is not performed, multi-layer liquid penetrant examination must be performed. Welds *other than the field closure welds* are also helium leak tested *in the fabrication shop*, providing added assurance of weld integrity. Additionally, a ~~hydrostatic~~ *Code pressure* test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity *after fuel loading*. The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld does not result in release of radioactive material to the environment. Table ~~7.1.3~~ *9.1.4* provides a summary of the closure weld examinations and tests.

### 7.1.4 Closure

The MPC is a totally seal-welded pressure vessel. The MPC has no bolted closure or mechanical seals. The MPC's redundant closures are designed to maintain confinement integrity during normal conditions of storage, and off-normal and postulated accident conditions. There are no unique or special closure devices. Primary closure welds (lid-to-shell and vent/drain port cover plate-to-lid) are examined ~~and leakage tested~~ *using the liquid penetrant technique* to ensure their integrity. A description of the MPC weld examinations is provided in Chapter 9.

Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required. ~~Section 11.2.1.4~~ *Chapter 11* describes requirements for verifying the continued confinement capabilities of the MPC in the event of off-normal or accident conditions. As discussed in Section 2.3.3.2, no instrumentation is required or provided for HI-STORM 100 storage operations, other than normal security service instruments and TLDs.

### 7.1.5 Damaged Fuel Container

The MPC is designed to allow for the storage of specified damaged fuel assemblies and fuel debris in a specially designed damaged fuel container (DFC). Fuel assemblies classified as damaged fuel or fuel debris as specified in ~~the Approved Contents Section of Appendix B to the CoC~~ *Section 2.1.9* have been evaluated.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel DFCs *for storage prior to placement* in the HI-STORM 100 System. The DFCs that may be loaded into the MPCs are *discussed in Section 2.1.3 shown in Figures 2.1.1 through Figure 2.1.2e*. The DFC is designed to provide SNF loose component retention and handling capabilities. The DFC consists of a smooth-walled, welded stainless steel square container with a removable lid. The container lid provides the means of DFC closure and handling. The DFC is provided with stainless steel wire mesh screens in the top and bottom for draining, moisture removal and helium backfill operations. The screens are specified as a 250-by-250-mesh with an effective opening of 0.0024 inches. There are no other openings in the DFC. *Section 2.1.9* ~~The CoC~~ specifies the fuel assembly characteristics for damaged fuel acceptable for loading in the MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F or MPC-68FF and for fuel debris acceptable for loading in the MPC-24EF, MPC-32F, MPC-68F or MPC-68FF.

Since the DFC has screens on the top and bottom, the DFC provides no pressure retention function. The confinement function of the DFC is limited to minimizing the release of loose particulates within the sealed MPC. The storage design basis leakage rates are not altered by the presence of the DFCs. The radioactive material available for release from the specified fuel assemblies are bounded by the design basis fuel assemblies analyzed herein.

#### *7.1.6 Design and Qualification of Final MPC Closure Welds*

*The Holtec MPC final closure welds meet the criteria of NRC Interim Staff Guidance (ISG)18 [7.1.2] such that leakage from the MPC confinement boundary is not considered credible. Table 7.1.4 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one. In addition, because proper execution of the MPC lid-to-shell weld is vital to ensuring leak-tightness of each field-welded MPC, Holtec shall review the closure welding procedures for conformance to Code and FSAR requirements.*

Table 7.1.1

SUMMARY OF CONFINEMENT BOUNDARY DESIGN SPECIFICATIONS

<b>Design Condition</b>	<b>Design Pressure (psig)</b>	<b>Design Temperature (°F)</b>
Normal	100	MPC Lid: 550
		MPC Shell: 50450
		MPC Baseplate: 400
Off-Normal	<del>110</del> 100	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775
Accident	200	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775

Table 7.1.2

MPC CONFINEMENT BOUNDARY WELDS

<b>Confinement Boundary Welds</b>		
<b>MPC Weld Location</b>	<b>Weld Type†</b>	<b>ASME Code Category (Section III, Subsection NB)</b>
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port cover plates to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to closure ring radial	Partial Penetration Groove (field weld)	††
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C

† The tests and inspections for the confinement boundary welds are listed in Section 9.1.1.

†† This joint is governed by NB-5271 (liquid penetrant examination).

Table 7.1.3

## CLOSURE WELD EXAMINATIONS AND TESTS

TABLE DELETED

Closure Weld Description	Inspections/Tests	ASME Acceptance Criteria
MPC Lid to Shell	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass Volumetric Examination of Weld (UT) or multi-layer PT Hydrostatic Test Post Hydrostatic Test—PT Helium Leakage Test	NF-5360 NB-5350 NB-5350 NF-5360 NB-5332  NB-6000 NB-5350 Sect. V and ANSI N14.5
Vent/Drain Cover Plate to MPC Lid	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass Helium Leakage Test	NF-5360 NB-5350 NB-5350 NF-5360 Sect. V and ANSI N14.5
Closure Ring Radial Welds	VT on Tack Welds PT Root Pass — (if multiple pass) PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360
Closure Ring to MPC Shell	VT on Tack Welds PT Root Pass — (if multiple pass) PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360
Closure Ring to MPC Lid	VT on Tack Welds PT Root Pass PT Final Pass VT Final Pass	NF-5360 NB-5350 NB-5350 NF-5360



Table 7.1.4

COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE FOR STORAGE

<b>DESIGN/QUALIFICATION GUIDANCE</b>	<b>HOLTEC MPC DESIGN</b>	<b>FSAR REFERENCE</b>
<i>The canister is constructed from austenitic stainless steel</i>	<i>The MPC enclosure vessel is constructed entirely from austenitic stainless steel (Alloy X). Alloy X is defined as Type 304, 304LN, 316, or 316LN material</i>	<i>Section 1.2.1.1 and Appendix 1.A</i>
<i>The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3</i>	<i>The MPC lid-to-shell (LTS) closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8” of weld depth, which corresponds to the critical flaw size. A weld quality factor of 0.45 (45% of actual weld capacity) has been used in the stress analysis.</i>	<i>Section 9.1.1.1 and Tables 2.2.15 and 9.1.4.  HI-STAR FSAR Section 3.4.4.3.1.5 and Appendix 3.E (Docket 72-1008)</i>
<i>The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents, and natural phenomena</i>	<i>The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is design to withstand 45 g deceleration loadings and the cask system is analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g’s.</i>	<i>Section 3.4.4.3 and Appendix 3.A.  HI-STAR FSAR Section 3.4.4.3</i>
<i>Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10 CFR 72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.</i>	<i>Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STORM CoC. Compliance with 10 CFR 72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.</i>	<i>Section 9.1.1.1 and Table 2.2.15  Section 13.0</i>
<i>Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.</i>	<i>The NRC has approved the Holtec quality assurance program under 10 CFR 71. That QA program approval has been adopted for activities governed by 10 CFR 72 as permitted by 10 CFR 72.140(d)</i>	<i>Section 13.0</i>

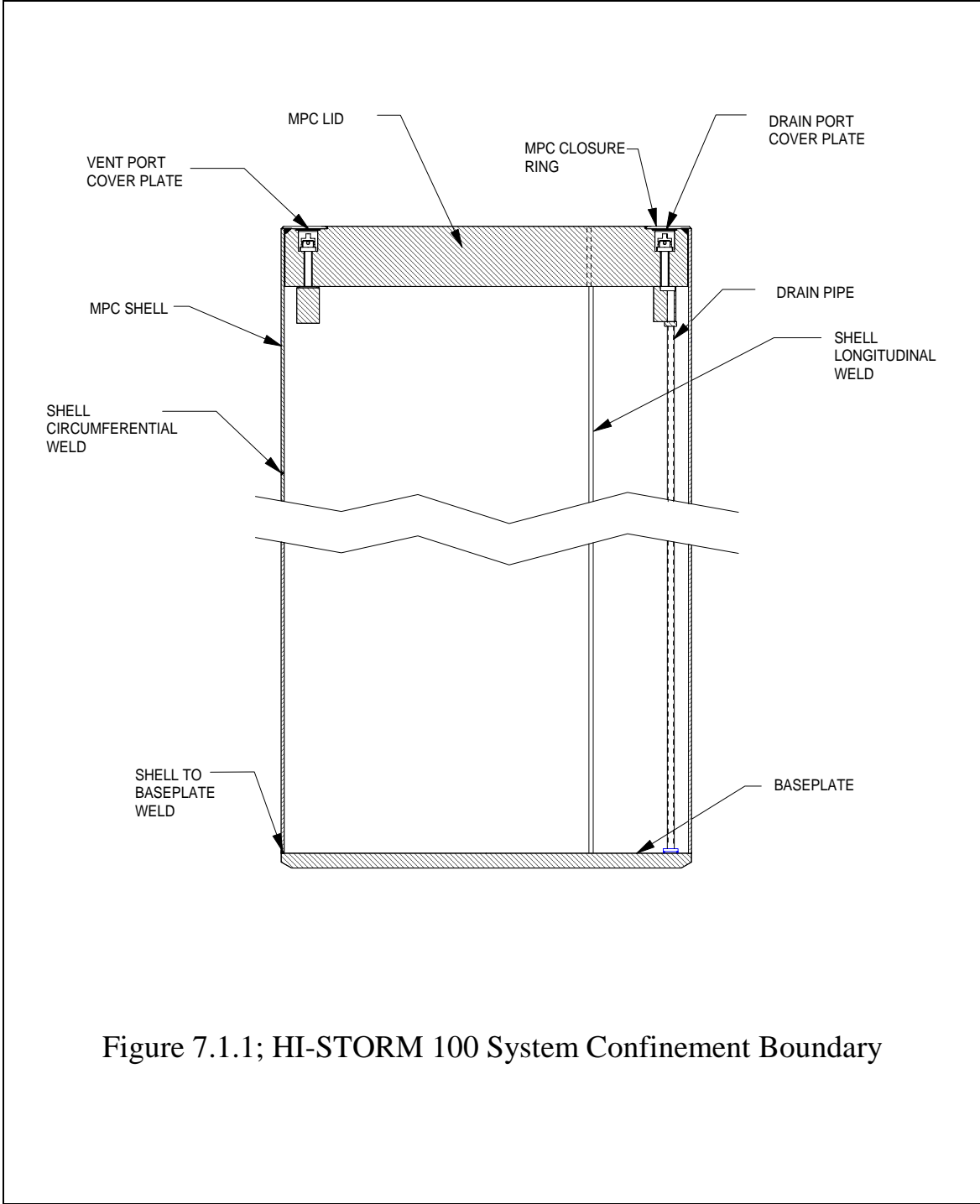


Figure 7.1.1; HI-STORM 100 System Confinement Boundary

REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

The MPC uses multiple confinement barriers provided by the fuel cladding and the MPC enclosure vessel to assure that there is no release of radioactive material to the environment. Chapter 3 shows that all confinement boundary components are maintained within their Code-allowable stress limits during normal *and off-normal* storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal *and off-normal* conditions of storage. *Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.* Since the MPC confinement vessel remains intact, and the design bases temperatures and pressure are not exceeded, *leakage from the MPC confinement boundary is not credible* ~~the design basis leakage rate is not exceeded~~ during normal *and off-normal* conditions of storage.

CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT  
CONDITIONS

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials, including fission gases, volatiles, fuel fines or crud, for postulated storage accident conditions. The analyses presented in Chapters 3 and 11 demonstrate that the MPC remains intact during all ~~normal, off-normal and~~ postulated accident conditions, including the associated increased internal pressure due to decay heat generated by the stored fuel. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], *with certain NRC-approved alternatives as listed in Table 2.2.15. to the maximum extent practicable.* Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. In summary, there is no mechanistic failure that results in a breach of, and associated leakage of radioactive material from the MPC confinement boundary.

## 7.4 REFERENCES

- [7.0.1] 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.
- [7.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January, 1997.
- [7.1.1] American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, Class 1 Components, 1995 Edition.
- [7.12.42] *Interim Staff Guidance 18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," May 2003. Interim Staff Guidance 11, Revision 1, "Transportation and Storage of Spent Fuel Having Burnups in Excess of 45GWD/MTU", May 16, 2000.*
- [7.2.2] ~~Interim Staff Guidance 5, Revision 1, "Normal, Off Normal, and Hypothetical Dose Estimate Calculations", June 18, 1999 Deleted.~~
- [7.3.1] Deleted.
- [7.3.2] ~~Anderson, B.L. et al. *Containment Analysis for Type B Packages Used to Transport Various Contents*. NUREG/CR-6487, UCRL ID-124822. Lawrence Livermore National Laboratory, November 1996 Deleted.~~
- [7.3.3] ~~Shleien, B, *The Health Physics and Radiological Health Handbook*, Scinta, Inc. Silver Spring, MD, 1992 Deleted.~~
- [7.3.4] ~~U.S. Nuclear Regulatory Commission, *Atmospheric Dispersement Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Regulatory Guide 1.145, February 1989 Deleted.~~
- [7.3.5] ~~U.S. EPA, Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, DE89-011065, 1988 Deleted.~~
- [7.3.6] ~~U.S. EPA, Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil*, EPA 402-R-93-081, 1993 Deleted.~~
- [7.3.7] ~~International Commission on Radiological Protection, *Limits for Intakes of Radionuclides by Workers*, ICRP Publication 30, Part 1; Pergamon Press; Oxford; 1978 Deleted.~~
- [7.3.8] ~~ANSI N14.5-1997. *American National Standard for Radioactive Material Leakage Tests*~~

~~on Packages for Shipment.~~*Deleted.*

[7.3.9] ~~Rosenhow, W.M. and Hartnett, J.P., *Handbook of Heat Transfer*, McGraw-Hill Book Company, New York, 1973~~*Deleted.*

[7.3.10] ~~Turner, James E. *Atoms, Radiation, and Radiation Protection*, McGraw-Hill Book Company, New York, 1992~~*Deleted.*

[7.3.11] ~~Rashid, Y.R., et al, "An Estimate of the Contribution of Spent Fuel Products to the Releaseable Source Term in Spent Fuel Transportation Casks," SAND88-2778C, Sandia National Laboratories, 1988~~  
*Deleted.*

APPENDIX 7.A

*DELETED*

~~EXAMPLE DOSE CALCULATIONS FOR NORMAL, OFF-NORMAL, AND  
ACCIDENT CONDITIONS OF STORAGE~~

~~MPC-32, Normal Conditions of Storage, Dose from Inhalation: 7 pages  
MPC-32, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages  
MPC-32, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-32, Normal Conditions of Storage, Dose from Submersion: 8 pages  
MPC-32, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages  
MPC-32, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Inhalation: 7 pages  
MPC-68, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages  
MPC-68, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Submersion: 8 pages  
MPC-68, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages  
MPC-68, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

## CHAPTER 8: OPERATING PROCEDURES<sup>†</sup>

### 8.0 INTRODUCTION:

This chapter outlines the loading, unloading, and recovery procedures for the HI-STORM 100 System for storage operations. The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance is met and the requirements of the CoC are met. The information provided in this chapter meets all requirements of NUREG-1536 [8.0.1].

Section 8.1 provides the guidance for loading the HI-STORM 100 System in the spent fuel pool. Section 8.2 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 8.3 provides the procedure for unloading the HI-STORM 100 System in the spent fuel pool. Section 8.4 provides the guidance for MPC transfer to the HI-STAR 100 Overpack for transport or storage. Section 8.4 can also be used for recovery of a breached MPC for transport or storage. Section 8.5 provides the guidance for transfer of the MPC into HI-STORM from the HI-STAR 100 transport overpack. ~~The Technical Specifications in Appendix A to CoC 72-1014 provide Limiting Conditions of Operation (LCO), Surveillance Requirements (SR's), as well as administrative information, such as Use and Application. Appendix B to COC 72-1014 provides the approved contents and design features applicable to the HI-STORM 100 System. FSAR Appendix 12.A includes the Bases for the LCOs. The Technical Specifications impose restrictions and requirements that must be applied throughout the loading and unloading process.~~ Equipment specific operating details such as Vacuum Drying System, valve manipulation and Transporter operation are not within the scope of this FSAR and will be provided to users based on the specific equipment selected by the users and the configuration of the site.

The procedures contained herein describe acceptable methods for performing HI-STORM 100 loading and unloading operations. Unless otherwise stated, references to the HI-STORM 100 apply equally to the HI-STORM 100 and the HI-STORM 100S. Users may alter these procedures to allow alternate methods and operations to be performed in parallel or out of sequence as long as the general intent of the procedure is met. In the figures following each section, acceptable configurations of rigging, piping, and instrumentation are shown. In some cases, the figures are artists renditions. Users may select alternate configurations, equipment and methodology to accommodate their specific needs provided that the intent of this guidance is met and the requirements of the CoC are met. All rigging should be approved by the user's load handling authority prior to use. User-developed procedures and the design and operation of any alternate equipment must be reviewed by the Certificate holder prior to implementation.

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



Licensees (Users) will utilize the procedures provided in this chapter, ~~the Technical Specifications in Appendix A to CoC 72-1014, the conditions of the Certificate of Compliance,~~ equipment-specific operating instructions, and plant working procedures and apply them to develop the site specific written, loading and unloading procedures.

The loading and unloading procedures in Section 8.1 and 8.3 can also be appropriately revised into written site-specific procedures to allow dry loading and unloading of the system in a hot cell or other remote handling facility. The Dry Transfer Facility (DTF) loading and unloading procedures are essentially the same with respect to loading, ~~and vacuum drying~~ *removing moisture*, inerting, and leakage testing of the MPC. The dry transfer facility shall develop the appropriate site-specific procedures as part of the DTF facility license.

Tables 8.1.1 through 8.1.4 provide the handling weights for each of the HI-STORM 100 System major components and the loads to be lifted during various phases of the operation of the HI-STORM 100 System. Users shall take appropriate actions to ensure that the lift weights do not exceed user-supplied lifting equipment rated loads. Table 8.1.5 provides the HI-STORM 100 System bolt torque and sequencing requirements. Table 8.1.6 provides an operational description of the HI-STORM 100 System ancillary equipment along with its safety designation, where applicable. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in the Certificate of Compliance and as defined in ~~the Appendix B to CoC 72-1014~~ *Section 2.1.9* are loaded into the HI-STORM 100 System.

In addition to the requirements set forth in the CoC, users will be required to develop or modify existing programs and procedures to account for the operation of an ISFSI. Written procedures will be required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of items and components identified as Important to Safety, 10CFR72.48 [8.1.1] programs, specialized instrument calibration, special nuclear material accountability at the ISFSI, security modifications, fuel handling procedures, training and emergency response, equipment and process qualifications. Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. *Users are also required to take necessary actions to prevent the fuel cladding from exceeding temperature limits during drying operations and during handling of the MPC in the HI-TRAC transfer cask. Section 4.5 of the FSAR provides requirements on the necessary actions, if any, based on the heat load of the MPC.*

Table 8.1.7 summarizes some of the instrumentation used to load and unload the HI-STORM 100 System. ~~Other instrumentation that meets the requirements of the Technical Specifications is also acceptable.~~ Tables 8.1.8, 8.1.9, and 8.1.10 provide sample receipt inspection checklists for the HI-STORM 100 overpack, the MPC, and the HI-TRAC Transfer Cask, respectively. Users may develop site-specific receipt inspection checklists, as required for their equipment. Fuel handling, including the handling of fuel assemblies in the Damaged Fuel Container (DFC) shall

be performed in accordance with written site-specific procedures. DFCs shall be loaded in the spent fuel pool racks prior to placement into the MPC.

### **Technical and Safety Basis for Loading and Unloading Procedures**

The procedures herein (Sections 8.1.2 through 8.1.5) are developed for the loading, storage, unloading, and recovery of spent fuel in the HI-STORM 100 System. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present risks. The design of the HI-STORM 100 System, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize risks and mitigate consequences of potential events. To summarize, consideration is given in the loading and unloading systems and procedures to the potential events listed in Table 8.0.1.

The primary objective is to reduce the risk of occurrence and/or to mitigate the consequences of the event. The procedures contain Notes, Warnings, and Cautions to notify the operators to upcoming situations and provide additional information as needed. The Notes, Warnings and Cautions are purposely bolded and boxed and immediately precede the applicable steps.

In the event of an extreme abnormal condition (e.g., cask drop or tip-over event) the user shall have appropriate procedural guidance to respond to the situation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution and reporting to the appropriate regulatory agencies, as required.

Table 8.0.1  
OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6. Procedural guidance is given for cask handling, inspection of lifting equipment, and proper engagement to the trunnions. <del>Technical Specifications limit the cask and overpack lift height outside the fuel building.</del>	See Section 8.1.2. <del>See Technical Specifications in Appendix A to CoC 72-1014 for HI-TRAC and HI-STORM lift height limitations.</del>
Cask Tip-Over Prior to welding of the MPC lid	The Lid Retention System is available to secure the MPC lid during movement between the spent fuel pool and the cask preparation area.	See Section 8.1.5. See Figure 8.1.15.
Contamination of the MPC external shell	The annulus seal, pool lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water. <del>Technical Specifications require surveys of certain components of the HI-STORM 100 System to monitor for removable contamination.</del>	See Figures 8.1.13 and 8.1.14. <del>See Technical Specifications in Appendix A to CoC 72-1014.</del>
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.	See Figures 8.1.19-8.1.22.
Damage to fuel assembly cladding from oxidation/thermal shock	Fuel assemblies are never subjected to air or oxygen during loading and unloading operations. Cool-Down System brings fuel assembly bulk temperatures to below water boiling temperature prior to flooding.	See Section 8.1.5, and Section 8.3.3 <del>and LCO 3.1.3.</del>
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum Drying System is separate from pressurized gas and water systems.	See Figure 8.1.22 and 8.1.23.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to, and during welding or cutting activities. It is recommended for defense-in-depth that the space below the MPC lid be evacuated or purged prior to, and during these activities.	See Section 8.1.5 and Section 8.3.3.

Table 8.0.1  
OPERATIONAL CONSIDERATIONS  
(CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Excess dose from failed fuel assemblies	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.	See Figure 8.1.16 and Section 8.3.3.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.	See ALARA Notes and Warnings throughout the procedures.
Excess generation of radioactive waste	The HI-STORM system uses process systems that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Where possible, items are installed by hand and require no tools.	Examples: HI-TRAC bottom protective cover, bolt plugs in empty holes, pre-wetting of components.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.	See Section 8.1.4.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in stages to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. <i>If the forced helium dehydration process is used, the temperature of the gas exiting the demister is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations.</i>	See Section 8.1.5, and <del>Technical Specification LCO 3.1.1, and Appendix 12.A, Bases B 3.1.1.</del>
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.	See Section 8.1.5.

Table 8.0.1  
OPERATIONAL CONSIDERATIONS  
(CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Load Drop	Rigging diagrams and procedural guidance are provided for all lifts. Component weights are provided in Tables 8.1.1 through 8.1.4.	See Figures 8.1.6, 8.1.7, 8.1.9, 8.1.25 and 8.1.27. See Tables 8.1.1 through 8.1.4.
Over-pressurization of MPC during loading and unloading	Pressure relief valves in the water and gas processing systems limit the MPC pressure to acceptable levels.	See Figures 8.1.20, 8.1.21, 8.1.23 and 8.3.4.
Overstressing MPC lift lugs from side loading	The MPC is upended using the upending frame. The lift lugs are never side loaded.	See Figure 8.1.6 and Section 8.1.2.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts.	See Section 8.1.7 for example. See Tables 8.1.1 through 8.1.4.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.	See Section 8.1.5 and Section 8.3.3.
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.	See Section 8.1.3 and Section 8.1.5.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during <i>moisture removal</i> , <del>vacuum drying</del> and <i>helium</i> backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.	See Figure 8.1.11 and 8.1.16. See Section 8.3.3.

## 8.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

### 8.1.1 Overview of Loading Operations:

The HI-STORM 100 System is used to load, transfer and store spent fuel. Specific steps are performed to prepare the HI-STORM 100 System for fuel loading, to load the fuel, to prepare the system for storage and to place it in storage at an ISFSI. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC and/or HI-STORM may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications as long as the ~~Technical Specification~~ lift height restrictions are met (lift height restrictions apply only to suspended forms of transport). Users shall develop detailed written procedures to control on-site transport operations. Section 8.1.2 provides the general procedures for rigging and handling of the HI-STORM overpack and HI-TRAC transfer cask. Figure 8.1.1 shows a general flow diagram of the HI-STORM loading operations.

Refer to the boxes of Figure 8.1.2 for the following description. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into HI-TRAC (Box 2). The annulus is filled with plant demineralized water<sup>†</sup> and the MPC is filled with either spent fuel pool water or plant demineralized water (*borated as required*) (Box 3). An inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC to prevent spent fuel pool water from contaminating the exterior surface of the MPC. HI-TRAC and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the optional Lid Retention System (Box 6). The lift yoke remotely engages to the HI-TRAC lifting trunnions to lift the HI-TRAC and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC from the spent fuel pool, the cask is removed from the spent fuel pool. If the Lid Retention System is being used, the HI-TRAC top lid bolts are installed to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

HI-TRAC is placed in the designated preparation area and the Lift Yoke and Lid Retention System (if utilized) are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The Temporary Shield Ring (if utilized) is installed and filled with water and the neutron shield jacket is filled with water (if drained). The inflatable annulus seal is removed, and the annulus shield (if utilized) is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-TRAC during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped

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<sup>†</sup> Users may substitute domestic water in each step where demineralized water is specified.

into the annulus. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the automated welding system (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. An ultrasonic or multi-layer PT examination is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The water level is raised to the top of the MPC and a hydrostatic test followed by an additional liquid penetrant examination is performed on the MPC Lid-to-Shell weld to verify structural integrity. ~~A small amount of water is displaced with helium gas for leakage testing. A leakage rate test is performed on the MPC lid to shell weld to verify weld integrity and to ensure that leakage rates are within acceptance criteria (See Technical Specification LCO 3.1.1).~~

To calculate the helium backfill requirements for the MPC, the free volume inside the MPC must first be determined. This free volume may be determined by measuring the volume of water displaced or any other suitable means.

*Depending upon the burn-up of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system or forced helium dehydration system. For MPCs without high burn-up fuel, ~~t~~The vacuum drying system ~~is~~may be connected to the MPC and is used to remove all liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and vacuum drying system lines. The internal pressure is reduced to below 3 torr and held for 30 minutes to ensure that all liquid water is removed ~~(See Technical Specification LCO 3.1.1).~~*

*~~Alternatively f~~For high burn-up fuel, or as an alternative for MPCs without high burn-up fuel, a forced helium dehydration ~~moisture removal~~ system is utilized to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. The temperature of the gas exiting the system ~~demoisturizer~~ is maintained below 21 °F for a minimum of 30 minutes to ensure that all liquid water is removed ~~Gas exiting the MPC is monitored for entrained moisture until no discernable moisture is present in the MPC.~~*

~~Following MPC drying ~~moisture removal~~, the MPC is evacuated and backfilled with a predetermined ~~pressure amount~~ of helium gas ~~(See Technical Specification LCO 3.1.1).~~ If the MPC contains high burn-up fuel, then a Supplemental Cooling System (SCS) (if required) is connected to the HI-TRAC annulus prior to helium backfill and is used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits (See Figure 2.C.1). ~~Limitations for the at vacuum duration are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded although a time limit of less than 2 hours at vacuum will bound any MPC.~~The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides ~~the a means of~~ for future leakage rate testing of the MPC confinement boundary welds. Cover plates are installed and seal welded over the MPC vent and drain ports with liquid penetrant examinations performed on the~~

root and final passes (for multi-pass welds) (Box 10). ~~The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.~~

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield (if utilized) is removed and the remaining water in the annulus is drained. The Temporary Shield Ring (if utilized) is drained and removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination (~~See Technical Specification LCO 3.2.2~~) and HI-TRAC dose rates are measured. HI-TRAC top lid<sup>3</sup> is installed and the bolts are torqued (Box 11). The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point on the MPC. MPC slings are installed between the MPC lift cleats and the lift yoke (Box 12).

If the HI-TRAC 125 is not being used, the transfer lid is attached to the HI-TRAC as follows. The HI-TRAC is positioned above the transfer slide to prepare for bottom lid replacement. The transfer slide consists of an adjustable-height rolling carriage and a pair of channel tracks. The transfer slide supports the transfer step which is used to position the two lids at the same elevation and creates a tight seam between the two lids to eliminate radiation streaming. The overhead crane is shut down to prevent inadvertent operation. The transfer slide carriage is raised to support the pool lid while the bottom lid bolts are removed. The transfer slide then lowers the pool lid and replaces the pool lid with the transfer lid. The carriage is raised and the bottom lid bolts are replaced. The MPC lift cleats and slings support the MPC during the transfer operations. Following the transfer, the MPC slings are disconnected and HI-TRAC is positioned for MPC transfer into HI-STORM.

MPC transfer may be performed inside or outside the fuel building (Box 13). Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways (Box 14 and 15). The empty HI-STORM overpack is inspected and positioned with the lid removed. Vent duct shield inserts<sup>1</sup> are installed in the HI-STORM exit vent ducts. The vent duct shield inserts prevent radiation streaming from the HI-STORM Overpack as the MPC is lowered past the exit vents. If the HI-TRAC 125D is used, the mating device is positioned on top of the HI-STORM. The HI-TRAC is placed on top of HI-STORM. An alignment device (or mating device in the case of HI-TRAC 125D) helps guide HI-TRAC during this operation<sup>2</sup>. The MPC may be lowered using the MPC downloader, the main crane hook or other similar devices. The MPC downloader (if used) may be attached to the HI-TRAC lid or mounted to the overhead lifting device. The MPC slings are attached to the MPC lift cleats.

*If used, the SCS will be disconnected from the HI-TRAC and the HI-TRAC annulus drained, prior to transfer of the MPC from the HI-TRAC to the HI-STORM.* If the transfer doors are used

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<sup>1</sup> Vent duct shield inserts are only used on the HI-STORM 100.

<sup>2</sup> The alignment guide may be configured in many different ways to accommodate the specific sites. See Table 8.1.6.

<sup>3</sup> Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid



(i.e. not the HI-TRAC 125D), the MPC is raised slightly, the transfer lid door locking pins are removed and the doors are opened. If the HI-TRAC 125D is used, the pool lid is removed and the mating device drawer is opened. Optional trim plates may be installed on the top and bottom of both doors (or drawer for HI-TRAC 125D) and secured using hand clamps. The trim plates eliminate radiation streaming above and below the doors (drawer). The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. The trim plates are removed, the doors (or drawer) are closed. The empty HI-TRAC must be removed with the doors open when the HI-STORM 100S is used to prevent interference with the lift cleats and slings. HI-TRAC is removed from on top of HI-STORM. The MPC slings and MPC lift cleats are removed. Hole plugs are installed in the empty MPC lifting holes to fill the voids left by the lift cleat bolts. The alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (if used) are removed, and the HI-STORM lid is installed. The exit vent gamma shield cross plates temperature elements (if used) and vent screens are installed. The HI-STORM lid studs and nuts are installed. The HI-STORM is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM Overpack and HI-TRAC transfer cask may be moved using a number of methods as long as the lifting equipment requirements ~~in the Technical Specification~~ are met. For sites with high seismic conditions, the HI-STORM 100A is anchored to the ISFSI. Once located at the storage pad, the inlet vent gamma shield cross plates are installed and the shielding effectiveness test is performed. Finally, the temperature elements and their instrument connections are installed (if used), and the air temperature rise testing (if required ~~by the Technical Specifications~~) is performed to ensure that the system is functioning within its design parameters.

#### 8.1.2 HI-TRAC and HI-STORM Receiving and Handling Operations

**Note:**

HI-TRAC may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for HI-TRAC and HI-STORM handling. Site-specific procedures shall specify the required operational sequences based on the handling configuration at the sites. ~~Refer to the Technical Specifications for loaded HI-TRAC and HI-STORM 100 Overpack handling limitations.~~

1. Vertical Handling of HI-TRAC:
  - a. Verify that the lift yoke load test certifications are current.
  - b. Visually inspect the lifting device (lift yoke or lift links) and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Replace or repair damaged components as necessary.
  - c. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
  - d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

**Note:**

Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements. ~~Refer to Technical Specification 4.9 for additional equipment handling requirements.~~

**Warning:**

When lifting the loaded HI-TRAC with only the pool lid, the HI-TRAC should be carried as low as practicable. This minimizes the dose rates due to radiation scattering from the floor. Personnel should remain clear of the area and the HI-TRAC should be placed in position as soon as practicable.

- e. Raise HI-TRAC and position it accordingly.
2. Upending of HI-TRAC in the Transfer Frame:
- a. Position HI-TRAC under the lifting device. Refer to Step 1, above.
  - b. If necessary, remove the missile shield from the HI-TRAC Transfer Frame. See Figure 8.1.4.
  - c. Verify that the lift yoke load test certifications are current.
  - d. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
  - e. Deleted.
  - f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
  - g. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.
  - h. Slowly rotate HI-TRAC to the vertical position keeping all rigging as close to vertical as practicable. See Figure 8.1.4.
  - i. If used, lift the pocket trunnions clear of the Transfer Frame rotation trunnions.
3. Downending of HI-TRAC in the Transfer Frame:

**ALARA Warning:**

A loaded HI-TRAC should only be downended with the transfer lid or other auxiliary shielding installed.

- a. Position the Transfer Frame under the lifting device.
- b. Verify that the lift yoke load test certifications are current.
- c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
- d. Deleted.
- e. Deleted.

- f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
  - g. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.
  - h. Position the pocket trunnions to receive the Transfer Frame rotation trunnions. See Figure 8.1.4 (Not used for HI-TRAC 125D).
  - i. Slowly rotate HI-TRAC to the horizontal position keeping all rigging as close to vertical as practicable.
  - j. Disengage the lift yoke.
4. Horizontal Handling of HI-TRAC in the Transfer Frame:
- a. Verify that the Transfer Frame is secured to the transport vehicle as necessary.
  - b. Downend HI-TRAC on the Transfer Frame per Step 3, if necessary.
  - c. If necessary, install the HI-TRAC missile Shield on the HI-STAR 100 Transfer Frame (See Figure 8.1.4).
5. Vertical Handling of HI-STORM:

**Note:**

The HI-STORM 100 Overpack may be lifted with a special lifting device that engages the overpack anchor blocks with threaded studs and connects to a cask transporter, crane, or similar equipment. The device is designed in accordance with ANSI N14.6.

- a. Visually inspect the HI-STORM lifting device for gouges, cracks, deformation or other indications of damage.
  - b. Visually inspect the transporter lifting attachments for gouges, cracks, deformation or other indications of damage..
  - c. If necessary, attach the transporter's lifting device to the transporter and HI-STORM..
  - d. Raise and position HI-STORM accordingly. See Figure 8.1.5.
6. Empty MPC Installation in HI-TRAC:

**Note:**

To avoid side loading the MPC lift lugs, the MPC must be upended in the MPC Upending Frame (or equivalent). See Figure 8.1.6.

- a. If necessary, rinse off any road dirt with water. Remove any foreign objects from the MPC internals.
- b. If necessary, upend the MPC as follows:
  - 1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.
  - 2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 8.1.6.

3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 8.1.6.
4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device) (See Figure 8.1.6).
5. Raise the MPC in the Upending Frame.

**Warning:**

The Upending Frame corner should be kept close to the ground during the upending process.

6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.
  7. When the MPC approaches the vertical orientation, tension on the lower slings may be released.
  8. Place the MPC in a vertical orientation.
  9. Disconnect the MPC straps and disconnect the rigging.
- c. Install the MPC in HI-TRAC as follows:
1. Install the four point lift sling to the lift lugs inside the MPC. See Figure 8.1.7.
  2. Raise and place the MPC inside HI-TRAC.

**Note:**

An alignment punch mark is provided on HI-TRAC and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring. See Figure 8.1.8.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside HI-TRAC. Disconnect the MPC rigging or the MPC lift rig.

8.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

**Note:**

Receipt inspection, installation of the empty MPC in the HI-TRAC, and lower fuel spacer installation may occur at any location or be performed at any time prior to complete submersion in the spent fuel pool as long as appropriate steps are taken to prevent contaminating the exterior of the MPC or interior of the HI-TRAC.

**ALARA Note:**

A bottom protective cover may be attached to HI-TRAC pool lid bottom. This will help prevent imbedding contaminated particles in HI-TRAC bottom surface and ease the decontamination effort.

1. Place HI-TRAC in the cask receiving area. Perform appropriate contamination and security surveillances, as required.
2. If necessary, remove HI-TRAC Top Lid by removing the top lid bolts and using the lift sling. See Figure 8.1.9 for rigging.
  - a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.
  - b. Perform a radiological survey of the inside of HI-TRAC to verify there is no residual contamination from previous uses of the cask.
3. Disconnect the rigging.
4. Store the Top Lid and bolts in a site-approved location.
5. If necessary, configure HI-TRAC with the pool lid as follows:

**ALARA Warning:**

The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

- a. Inspect the seal on the pool lid for cuts, cracks, gaps and general condition. Replace the seal if necessary.
  - b. Remove the bottom lid bolts and store them temporarily.
  - c. Raise the empty HI-TRAC and position it on top of the pool lid.
  - d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - e. Install the pool lid bolts. See Table 8.1.5 for torque requirements.
  - f. If necessary, thread the drain connector pipe to the pool lid.
  - g. Store the HI-TRAC Transfer Lid in a site-approved location.
6. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.
  7. Install the MPC inside HI-TRAC and place HI-TRAC in the designated preparation area. See Section 8.1.2.

**Note:**

Upper fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Upper fuel spacer installation may occur any time prior to MPC lid installation.

8. Install the upper fuel spacers in the MPC lid as follows:

**Warning:**

Never work under a suspended load.

- a. Position the MPC lid on supports to allow access to the underside of the MPC lid.

- b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Figure 8.1.10 and Table 8.1.5 for torque requirements.
- c. Install threaded plugs in the MPC lid where and when spacers will not be installed, if necessary. See Table 8.1.5 for torque requirements.

9. At the user's discretion perform an MPC lid and closure ring fit test:

**Note:**

It may be necessary to perform the MPC installation and inspection in a location that has sufficient crane clearance to perform the operation.

- a. Visually inspect the MPC lid rigging (See Figure 8.1.9).
- b. At the user's discretion, raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid. Ensure that the reducer is fully seated against the bottom of the MPC lid. See Figure 8.1.11.
- c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. See Figure 8.1.12. Install the MPC lid. Verify that the MPC lid fit and weld prep are in accordance with the design drawings.

**ALARA Note:**

The closure ring is installed by hand. Some grinding may be required on the closure ring to adjust the fit.

- d. Install, align and fit-up the closure ring.
- e. Verify that closure ring fit and weld prep are in accordance with the fabrication drawings or the approved design drawings.
- f. Remove the closure ring, vent and drain port cover plates and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.

10. At the user's discretion, perform an MPC vent and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

**Note:**

Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used.

11. Install lower fuel spacers in the MPC (if necessary). See Figure 8.1.10.

12. Fill the MPC and annulus as follows:

- a. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

**Caution:**

Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

- b. Manually insert the inflatable annulus seal around the MPC. See Figure 8.1.13.

- c. Ensure that the seal is uniformly positioned in the annulus area.
- d. Inflate the seal.
- e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

**ALARA Note:**

Bolt plugs, placed in, or waterproof tape over empty bolt holes, reduce the time required for decontamination.

- 13. At the user's discretion, install HI-TRAC top lid bolt plugs and/or apply waterproof tape over any empty bolt holes.

**ALARA Note:**

Keeping the water level below the top of the MPC prevents splashing during handling.

- 14. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches below the top of the MPC shell. Refer to ~~LCO 3.3.1~~ *Tables 2.1.14 and 2.1.16* for boron concentration requirements.
- 15. If necessary for plant crane capacity limitations, drain the water from the neutron shield jacket. See Tables 8.1.1 through 8.1.4 as applicable.
- 16. Place HI-TRAC in the spent fuel pool as follows:

**ALARA Note:**

The term "Spent Fuel Pool" is used generically to refer to the users designated cask loading location. The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

- a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-TRAC. See Figure 8.1.14.
- b. Verify spent fuel pool for boron concentration requirements in accordance with ~~LCO 3.3.1~~ *Tables 2.1.14 and 2.1.16*.
- c. Engage the lift yoke to HI-TRAC lifting trunnions and position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

**ALARA Note:**

Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- d. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.
- e. When the top of the HI-TRAC reaches the elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

- f. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged. Remove the lift yoke from the spent fuel pool while spraying the crane cables and yoke with plant demineralized water.
- g. Observe the annulus seal for signs of air leakage. If leakage is observed (by the steady flow of bubbles emanating from one or more discrete locations) then immediately remove the HI-TRAC from the spent fuel pool and repair or replace the seal.

8.1.4 MPC Fuel Loading

<b>Note:</b>
An underwater camera or other suitable viewing device may be used for monitoring underwater operations.
<b>Note:</b>
<i>When loading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.</i>

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading as specified in ~~Appendix B to CoC 72-1014~~Section 2.1.9 have been selected for loading into the MPC.
2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.
3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.



### 8.1.5 MPC Closure

**Note:**

The user may elect to use the Lid Retention System (See Figure 8.1.15) to assist in the installation of the MPC lid and lift yoke, and to provide the means to secure the MPC lid in the event of a drop accident during loaded cask handling operations outside of the spent fuel pool. The user is responsible for evaluating the additional weight imposed on the cask, lift yoke, crane and floor prior to use. See Tables 8.1.1 through 8.1.4 as applicable. The following guidance describes installation of the MPC lid using the lift yoke. The MPC lid may also be installed separately.

Depending on facility configuration, users may elect to perform MPC closure operations with the HI-TRAC partially submerged in the spent fuel pool. If opted, operations involving removal of the HI-TRAC from the spent fuel pool shall be sequenced accordingly.

1. Remove the HI-TRAC from the spent fuel pool as follows:
  - a. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.
  - b. Install the drain line to the underside of the MPC lid. Ensure that the reducer is fully seated against the bottom of the MPC lid. See Figure 8.1.17.
  - c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage. See Figure 8.1.11 and 8.1.17.

**ALARA Note:**

Pre-wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- d. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented. See Figure 8.1.12.
- e. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

**Note:**

The outer diameter of the MPC lid will seat flush with the top edge of the MPC shell when properly installed.

- f. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.
- g. Engage the lift yoke to HI-TRAC lifting trunnions.

- h. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

**ALARA Note:**

Activated debris may have settled on the top face of HI-TRAC and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. Users are responsible for any water dilution considerations.

- i. Raise HI-TRAC until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Remove any activated or highly radioactive particles from HI-TRAC or MPC.
- j. Visually verify that the MPC lid is properly seated. Lower HI-TRAC, reinstall the lid, and repeat as necessary.
- k. Install the Lid Retention System bolts if the lid retention system is used.
- l. Continue to raise the HI-TRAC under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-TRAC reaches the same elevation as the reservoir, close the Annulus Overpressure System reservoir valve (if used). See Figure 8.1.14.

**Caution:**

Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action. One course of action involves initiating an MPC water flush for a certain duration and flow rate. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded. Refer to ~~LCO 3.3.1~~ Tables 2.1.14 and 2.1.16 for boron concentration requirements whenever water is added to the loaded MPC.

- m. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

**ALARA Note:**

Decontamination of HI-TRAC bottom should be performed using remote cleaning methods, covering or other methods to minimize personnel exposure. The bottom lid decontamination may be deferred to a convenient and practical time and location. Any initial decontamination should only be sufficient to preclude spread of contamination within the fuel building.

- n. Decontaminate HI-TRAC bottom and HI-TRAC exterior surfaces including the pool lid bottom. Remove the bottom protective cover, if used.
- o. If used, disconnect the Annulus Overpressure System from the HI-TRAC See Figure 8.1.14.
- p. Set HI-TRAC in the designated cask preparation area.

**Note:**

If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Users shall evaluate the cask weights to ensure that cask trunnion, lifting devices and equipment load limitations are not exceeded.

- q. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.
- r. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

**Warning:**

MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

- s. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.
- t. Perform decontamination and a dose rate/contamination survey of HI-TRAC.
- u. Prepare the MPC annulus for MPC lid welding as follows:

**ALARA Note:**

If the Temporary Shield Ring is not used, some form of gamma shielding (e.g., lead bricks or blankets) should be placed in the trunnion recess areas of the HI-TRAC water jacket to eliminate the localized hot spot.

- v. Decontaminate the area around the HI-TRAC top flange and install the Temporary Shield Ring, (if used). See Figure 8.1.18.

**ALARA Note:**

The water in the HI-TRAC-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

- æ-w. Attach the drain line to the HI-TRAC drain port and lower the annulus water level approximately 6 inches.
2. Prepare for MPC lid welding as follows:

**Note:**

The following steps use two identical Removable Valve Operating Assemblies (RVOAs) (See Figure 8.1.16) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

**Note:**

The vent and drain ports are opened by pushing the RVOA handle down to engage the square nut on the cap and turning the handle fully in the counter-clockwise direction. The handle will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the handle fully in the clockwise direction. The ports are closed when the handle cannot be turned further.

**Note:**

Steps involving preparation for welding may occur in parallel as long as precautions are taken to prevent contamination of the annulus.

- a. Clean the vent and drain ports to remove any dirt. Install the RVOAs (See Figure 8.1.16) to the vent and drain ports leaving caps open.

**ALARA Warning:**

Personnel should remain clear of the drain hoses any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

- b. Attach the water pump to the drain port (See Figure 8.1.19) and lower the water level to keep moisture away from the weld region.
- c. Disconnect the water pump.
- d. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable seal
- e. Deflate and remove the inflatable annulus seal.

**ALARA Note:**

The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontaminated.

- f. Survey the MPC lid top surfaces and the accessible areas of the top three inches of the MPC shell in accordance with the requirements of Technical Specification LCO 3.2.2.

**ALARA Note:**

The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

- g. Install the annulus shield. See Figure 8.1.13.

3. Weld the MPC lid as follows:

**ALARA Warning:**

Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

**ALARA Warning:**

It may be necessary to rotate or reposition the MPC lid slightly to achieve uniform weld gap and lid alignment. A punch mark is located on the outer edge of the MPC lid and shell. These marks are aligned with the alignment mark on the top edge of the HI-TRAC Transfer Cask (See Figure 8.1.8). If necessary, the MPC lid lift should be performed using a hand operated chain fall to closely control the lift to allow rotation and repositioning by hand. If the chain fall is hung from the crane hook, the crane should be tagged out of service to prevent inadvertent use during this operation. Continuous radiation monitoring is recommended.

- a. If necessary center the lid in the MPC shell using a hand-operated chain fall.

**Note:**

The MPC is equipped with lid shims that serve to close the gap in the joint for MPC lid closure weld.

- b. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

**ALARA Note:**

The AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

- c. Install the Automated Welding System baseplate shield. See Figure 8.1.9 for rigging.  
d. If used, install the Automated Welding System Robot.

**Note:**

It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.

**Caution:**

Oxidation of Boral panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid welding operations. *The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space.* ~~It is also recommended for defense in depth that the space below the MPC lid be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space.~~

- e. Perform combustible gas monitoring and, ~~if desired,~~ exhaust or purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.
  - f. Perform the MPC lid-to-shell weld and NDE with approved procedures (See 9.1 and Table 2.2.15).
  - g. Deleted.
  - h. Deleted.
  - i. Deleted.
  - j. Deleted.
4. Perform hydrostatic ~~and MPC leakage rate~~ testing as follows:

**ALARA Note:**

~~The leakage rates are determined~~ *Testing is performed* before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

- a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20 for the hydrostatic test arrangement.

**ALARA Warning:**

Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

- b. Fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose. Refer to ~~LCO~~ ~~3.3.1~~ *Tables 2.1.14 and 2.1.16* for boron concentration requirements.
- c. Perform a hydrostatic test of the MPC as follows:
  - 1. Close the drain valve and pressurize the MPC to 125 +5/-0 psig.
  - 2. Close the inlet valve and monitor the pressure for a minimum of 10 minutes. The pressure shall not drop during the performance of the test.

3. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water. The acceptance criteria is no observable water leakage.
  - d. Release the MPC internal pressure, disconnect the water fill line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.
    1. Repeat the liquid penetrant examination on the MPC lid final pass.
  - ~~e. Attach a regulated helium supply to the vent port and attach the drain line to the drain port as shown on Figure 8.1.21.~~
  - ~~f. Verify the correct pressure on the helium supply and open the helium supply valve. Drain approximately twenty gallons.~~
  - ~~g. Close the drain port valve and pressurize the MPC.~~
  - ~~h. Close the vent port.~~
  - ~~i. Perform a helium sniffer probe leakage rate test of the MPC lid to shell weld in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [8.1.2]. The MPC Helium Leak Rate shall be  $< 5.0E-6$  atm cc/sec (He) based on a 1 atmosphere pressure differential across the weld joint.~~
  - ~~j.e.~~ Repair any weld defects in accordance with the site's approved weld repair procedures. Reperform the Ultrasonic (if necessary), PT, and Hydrostatic and Helium Leakage tests if weld repair is performed.
5. Drain the MPC as follows:
- a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20.

**ALARA Warning:**  
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

- b. Attach the water fill line to the drain port and fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the drain line.
- c. Disconnect the water fill and drain lines from the MPC leaving the vent port valve open to allow for thermal expansion of the MPC water.

**ALARA Warning:**  
Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

- d. Attach a regulated helium or nitrogen supply to the vent port.
- e. Attach a drain line to the drain port shown on Figure 8.1.21.
- f. Deleted

- g. Verify the correct pressure on the gas supply.
- h. Open the gas supply valve and record the time at the start of MPC draining.

**Note:**

An optional warming pad may be placed under the HI-TRAC Transfer Cask to replace the heat lost during the evaporation process of MPC drying. This may be used at the user's discretion for older and colder fuel assemblies to reduce vacuum drying times.

- i. Start the warming pad, if used.

**Note:**

Users may continue to purge the MPC to remove as much water as possible.

- j. Drain the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve. See Figure 8.1.21.
- k. Deleted.
- l. Disconnect the gas supply line from the MPC.
- m. Disconnect the drain line from the MPC.

6. Dry the MPC as follows:

**Note:**

Vacuum drying or *forced helium dehydration* ~~moisture removal~~ (for high burn-up fuel) is performed to remove moisture and oxidizing gasses from the MPC. This ensures a suitable environment for long-term storage of spent fuel assemblies and ensures that the MPC pressure remains within design limits. The vacuum drying process described herein reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC. The moisture removal process limits bulk MPC temperatures by continuously circulating gas through the MPC. Steps ~~6.b-8.1.22a~~ through h in Section 8.1.5 are used for vacuum drying. Steps ~~8.1.226i~~ through k in Section 8.1.5 are used for moisture removal.

- a. *If using the vacuum drying system, go to Section 8.1.5 Step 6.b. If using the forced helium dehydration system, go to Section 8.1.5 Step 6.i. Attach the drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a for the vacuum drying system and 8.1.22b for the moisture removal system. Other equipment configurations that achieve the same results may also be used.*

**Note:**

The vacuum drying system may be configured with an optional fore-line condenser. Other equipment configurations that achieve the same results may be used.

- b. *Attach the vacuum drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a Deleted.*
- c. Deleted.



- d. Deleted.

**Note:**

*To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.*

- e. *Open the VDS suction valve and reduce the MPC pressure to below 3 torr.*  
~~Deleted.~~
- f. *Shut the VDS valves and verify a stable MPC pressure on the vacuum gauge.*  
~~Open the VDS suction valve and reduce the MPC pressure to below 3 torr.~~

**Note:**

*The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the vacuum drying system, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.*

- g. *Perform the MPC drying pressure test.*  
~~Shut the VDS valves and verify a stable MPC pressure on the vacuum gage.~~
- h. *Perform the MPC drying pressure test in accordance with the technical specifications.*  
~~Proceed to Step 6.n of Section 8.1.5 if not using the forced helium dehydration system.~~
- i. *Attach the moisture removal forced helium dehydration system to the vent and drain port RVOAs. See Figure 8.1.22b.*
- j. *Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.*

**NoteCaution:**

*Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance. For MPCs containing high burn-up fuel, the Supplemental Cooling System (SCS) (if required) is used to prevent fuel cladding temperatures from exceeding ISG-11 Rev. 3 limits. The implementation of cooling using the SCS will typically begin immediately following cessation of the drying operations with the FHD. Staging and check-out of the SCS shall be completed prior to the end of FHD operations to minimize the time to begin its operation.*

- k. *Continue the monitoring and moisture removal until LCO 3.1.1 is the acceptance criteria are met for MPC dryness.*
- l. *If required, connect the Supplemental Cooling System (SCS) to the HI-TRAC annulus (See Figure 2.C.1 for an example configuration).*

**Caution:**

*When water is first introduced to the HI-TRAC annulus for supplemental cooling, it will flash to steam until the MPC shell temperature has dropped to below the boiling point of the water. Appropriate precautions shall be in place prior to filling the annulus to prevent personal injury and damage to plant systems from the escaping water vapor.*

- m. If required, begin circulating the coolant through the HI-TRAC annulus to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits. Coolant flow rates shall be controlled to maintain adequate cooling in accordance with the thermal evaluation.*
  - n. Discontinue MPC drying operations.*
  - l.o. If necessary, attach the vacuum pump to the MPC.*
  - m.p. Evacuate the MPC to below 10 torr.*
  - n.q. Close the vent and drain port valves.*
  - o.r. Disconnect the VDS from the MPC.*
  - p.s. Stop the warming pad, if used.*
  - q.t. Close the drain port RVOA cap and remove the drain port RVOA.*
7. Backfill the MPC as follows:

**Note:**

Helium backfill shall be ~~in accordance with the Technical Specification at~~ *performed using helium with 99.995% (minimum) purity. Other equipment configurations that achieve the same results may be used.*

- a. Set the helium bottle regulator pressure to the appropriate pressure.
- b. Purge the Helium Backfill System to remove oxygen from the lines.
- c. Attach the Helium Backfill System to the vent port as shown on Figure 8.1.23 and open the vent port.
- d. Slowly open the helium supply valve while monitoring the pressure rise in the MPC.
- e. Deleted
- f. Deleted
- g. Deleted

**Note:**

If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.

- h. Carefully backfill the MPC ~~in accordance with the technical specifications~~ *in accordance with Table 1.2.2.*
- i. Disconnect the helium backfill system from the MPC.

- j. Close the vent port RVOA and disconnect the vent port RVOA.
8. Weld the vent and drain port cover plates as follows:

**Note:**

The process provided herein may be modified to perform actions in parallel. ~~Users may perform the final PT on the circumferential and plug welds at the same time.~~

- a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.
- b. Place the cover plate over the vent port recess.
- c. *Weld the cover plate.*

~~e. Weld the cover plate and perform NDE with approved procedures (See 9.1 and Table 2.2.15)~~

**Note:**

*ASME Boiler and Pressure Vessel Code [8.1.3], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.*

- ~~f.d.~~ *Perform NDE on the cover plate with approved procedures (See 9.1 and Table 2.2.15)* Deleted.
- ~~g.e.~~ *Repair any weld defects in accordance with the site's approved code weld repair procedures.* Deleted.
- ~~h.f.~~ Deleted.
- ~~i.g.~~ Deleted.
- ~~j.h.~~ Deleted.
- ~~k.i.~~ Repeat for the drain port cover plate.

~~9. Perform a leakage test of the MPC vent and drain port cover plates as follows:~~

- ~~a.~~
- ~~b. Repair any weld defects in accordance with the site's approved code weld repair procedures. Re-perform the leakage test as required.~~

~~10.9.~~ Weld the MPC closure ring as follows:

**ALARA Note:**

The closure ring is installed by hand. No tools are required. Localized grinding to achieve the desired fit and weld prep are allowed.

- a. Install and align the closure ring. See Figure 8.1.8.

- b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE with approved procedures (See 9.1 and Table 2.2.15).
- c. Deleted.
- d. Deleted.
- e. Deleted.
- f. Deleted.
- g. Deleted.
- h. Deleted.
- i. Deleted.
- j. If necessary, remove the AWS. See Figure 8.1.7 for rigging.

8.1.6 Preparation for Storage

<b>ALARA Warning:</b>
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.
<b>Caution:</b>
<i>Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.</i>

1. Remove the annulus shield (if used) and store it in an approved plant storage location
2. *If use of the SCS is not required, Attach a drain line to the HI-TRAC and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system.*
3. Install HI-TRAC top lid as follows:

<b>Warning:</b>
When traversing the MPC with the HI-TRAC top lid using non-single-failure proof (or equivalent safety factors), the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

- a. Install HI-TRAC top lid. Inspect the bolts for general condition. Replace worn or damaged bolts with new bolts.
  - b. Install and torque the top lid bolts. See Table 8.1.5 for torque requirements.
  - c. Inspect the lift cleat bolts for general condition. Replace worn or damaged bolts with new bolts.
  - d. Install the MPC lift cleats and MPC slings. See Figure 8.1.24 and 8.1.25. See Table 8.1.5 for torque requirements.
  - e. Drain and remove the Temporary Shield Ring, if used.
4. Replace the pool lid with the transfer lid as follows (Not required for HI-TRAC 125D):

**ALARA Note:**

The transfer slide is used to perform the bottom lid replacement and eliminate the possibility of directly exposing the bottom of the MPC. The transfer slide consists of the guide rails, rollers, transfer step and carriage. The transfer slide carriage and jacks are powered and operated by remote control. The carriage consists of short-stroke hydraulic jacks that raise the carriage to support the weight of the bottom lid. The transfer step produces a tight level seam between the transfer lid and the pool lid to minimize radiation streaming. The transfer slide jacks do not have sufficient lift capability to support the entire weight of the HI-TRAC. This was selected specifically to limit floor loads. Users should designate a specific area that has sufficient room and support for performing this operation.

**Note:**

The following steps are performed to pretension the MPC slings.

- a. Lower the lift yoke and attach the MPC slings to the lift yoke. See Figure 8.1.25.
- b. Raise the lift yoke and engage the lift yoke to the HI-TRAC lifting trunnions.
- c. If necessary, position the transfer step and transfer lid adjacent to one another on the transfer slide carriage. See Figure 8.1.26. See Figure 8.1.9 for transfer step rigging.
- d. Deleted.
- e. Position HI-TRAC with the pool lid centered over the transfer step approximately one inch above the transfer step.
- f. Raise the transfer slide carriage so the transfer step is supporting the pool lid bottom. Remove the bottom lid bolts and store them temporarily.

**ALARA Warning:**

Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

- g. Lower the transfer carriage and position the transfer lid under HI-TRAC.
- h. Raise the transfer slide carriage to place the transfer lid against the HI-TRAC bottom lid bolting flange.
- i. Inspect the transfer lid bolts for general condition. Replace worn or damaged bolts with new bolts.
- j. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.
- k. Raise and remove the HI-TRAC from the transfer slide.
- l. Disconnect the MPC slings and store them in an approved plant storage location.

**Note:**

HI-STORM receipt inspection and preparation may be performed independent of procedural sequence.

5. Perform a HI-STORM receipt inspection and cleanliness inspection in accordance with a site-approved inspection checklist, if required. See Figure 8.1.27 for HI-STORM lid rigging.

**Note:**

MPC transfer may be performed in the truck bay area, at the ISFSI, or any other location deemed appropriate by the licensee. The following steps describe the general transfer operations (See Figure 8.1.28). The HI-STORM may be positioned on an air pad, roller skid in the cask receiving area or at the ISFSI. The HI-STORM or HI-TRAC may be transferred to the ISFSI using a heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function (See Figure 8.1.29) as long as the HI-TRAC and HI-STORM lifting requirements as described in the Technical Specifications are not exceeded. The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

### 8.1.7 Placement of HI-STORM into Storage

1. Position an empty HI-STORM module at the designated MPC transfer location. The HI-STORM may be positioned on the ground, on a deenergized air pad, on a roller skid, on a flatbed trailer or other special device designed for such purposes. If necessary, remove the exit vent screens and gamma shield cross plates, temperature elements and the HI-STORM lid. See Figure 8.1.28 for some of the various MPC transfer options.
  - a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.
  - b. Transfer the HI-TRAC to the MPC transfer location.
2. De-energize the air pad or chock the vehicle wheels to prevent movement of the HI-STORM during MPC transfer and to maintain level, as required.

**ALARA Note:**

The HI-STORM vent duct shield inserts eliminate the streaming path created when the MPC is transferred past the exit vent ducts. Vent duct shield inserts are not used with the HI-STORM 100S.

3. Install the alignment device (or mating device for HI-TRAC 125D) and if necessary, install the HI-STORM vent duct shield inserts. See Figure 8.1.30.

**Caution:**

*For MPCs with high burn-up fuel requiring supplemental cooling, the time to complete the transfer may be limited to prevent fuel cladding temperatures in excess of ISG-11 Rev. 3 limits. (See Section 4.5) All preparatory work related to the transfer should be completed prior to terminating the supplemental cooling operations.*

4. ~~Deleted~~ If used, discontinue the supplemental cooling operations and disconnect the SCS. Drain the water from the HI-TRAC annulus to an appropriate plant discharge point.
5. Position HI-TRAC above HI-STORM. See Figure 8.1.28.

6. Align HI-TRAC over HI-STORM (See Figure 8.1.31) and mate the overpacks.
7. If necessary, attach the MPC Downloader. See Figure 8.1.32.
8. Attach the MPC slings to the MPC lift cleats.
9. Raise the MPC slightly to remove the weight of the MPC from the transfer lid doors (or pool lid for HI-TRAC 125D and mating device)
10. If using the HI-TRAC 125D, unbolt the pool lid from the HI-TRAC..
11. Remove the transfer lid door (or mating device drawer) locking pins and open the doors (or drawer).

**ALARA Warning:**

MPC trim plates are used to eliminate the streaming path above and below the doors (or drawer). If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

12. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
13. Lower the MPC into HI-STORM.
14. Disconnect the slings from the MPC lifting device and lower them onto the MPC lid.
15. Remove the trim plates (if used), and close the doors (or mating device drawer)

**ALARA Warning:**

Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed due to radiation streaming.

**Note:**

It may be necessary, due to site-specific circumstances, to move HI-STORM from under the empty HI-TRAC to install the HI-STORM lid, while inside the Part 50 facility. In these cases, users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

16. Remove HI-TRAC from on top of HI-STORM.
17. Remove the MPC lift cleats and MPC slings and install hole plugs in the empty MPC bolt holes. See Table 8.1.5 for torque requirements.
18. Place HI-STORM in storage as follows:
  - a. Remove the alignment device (mating device with HI-TRAC pool lid for HI-TRAC 125D) and vent duct shield inserts (if used). See Figure 8.1.30.
  - b. Inspect the HI-STORM lid studs and nuts for general condition. Replace worn or damaged components with new ones.

- c. If used, inspect the HI-STORM 100A anchor components for general condition. Replace worn or damaged components with new ones.
- d. Deleted.

**Warning:**

Unless the lift is single failure proof (or equivalent safety factor) for the HI-STORM Lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

**Note:**

Shims may be used on the HI-STORM 100 lid studs. If used, the shims shall be positioned to ensure a radial gap of less than 1/8 inch around each stud. The method of cask movement will determine the most effective sequence for vent screen, lid, temperature element, and vent gamma shield cross plate installation.

- e. Install the HI-STORM lid and the lid studs and nuts.. See Table 8.1.5 for bolting requirements. Install the HI-STORM 100 lid stud shims if necessary. See Figure 8.1.27 for rigging.
  - f. Install the HI-STORM exit vent gamma shield cross plates, temperature elements (if used) and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34a and 8.1.34b.
  - g. Remove the HI-STORM lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location. See Table 8.1.5 for torque requirements.
  - h. Secure HI-STORM to the transporter device as necessary.
19. Perform a transport route walkdown to ensure that the cask transport conditions are met. ~~See Technical Specification for the on-site cask handling limitations.~~
20. Transfer the HI-STORM to its designated storage location at the appropriate pitch. See Figure 8.1.35.

**Note:**

Any jacking system shall have the provisions to ensure uniform loading of all four jacks during the lifting operation.

- a. If air pads were used, insert the HI-STORM lifting jacks and raise HI-STORM. See Figure 8.1.36. Remove the air pad.
- b. Lower and remove the HI-STORM lifting jacks, if used.
- c. For HI-STORM 100A overpack (anchored), perform the following:
  - 1. Inspect the anchor stud receptacles and verify that they are clean and ready for receipt of the anchor hardware.
  - 2. Align the overpack over the anchor location.
  - 3. Lower the overpack to the ground while adjusting for alignment.



4. Install the anchor connecting hardware (See Table 8.1.5 for torque requirements).
21. Install the HI-STORM inlet vent gamma shield cross plates and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34.
22. Perform shielding effectiveness testing ~~per Technical Specification LCO 3.2.3.~~
23. Perform an air temperature rise test as follows for the first HI-STORM 100 System placed in service:

**Note:**

The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize. The purpose of this test is to confirm the initial performance of the HI-STORM 100 ventilation system.

- a. Measure the inlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average inlet air (or surface screen) temperature.
- b. Measure the outlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average outlet air (or surface screen) temperature.
- c. Determine the average air temperature rise by subtracting the results of the average inlet screen temperature from the average outlet screen temperature.
- d. Report the results to the certificate holder.

Table 8.1.1  
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS  
125-TON HI-TRAC\*\*

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case <sup>†</sup> Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid) <sup>††</sup>	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (232) overpack (without lid) <sup>††</sup>	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid) <sup>††</sup>	239,000	239,000	239,000					1	
HI-STORM 100S lid (without rigging)	25,500	25,500	25,500					1	
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,600	3,600	3,600	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Mating Device	15,000	15,000	15,000						
Empty HI-TRAC 125 (without Top Lid, neutron shield jacket water, or bottom lids)	117,803	117,803	117,803	1	1	1			1
Empty HI-TRAC 125D (without Top Lid, neutron shield jacket water, or bottom lids)	119,400	119,400	119,400	1	1	1			1
HI-TRAC 125 Top Lid	2,745	2,745	2,745			1			1
HI-TRAC 125D Top Lid	2,645	2,645	2,645			1			1
Optional HI-TRAC Lid Spacer (weight lbs/in thickness)	400	400	400						
HI-TRAC 125/125D Pool Lid(with bolts)	11,900	11,900	11,900	1	1				
HI-TRAC Transfer Lid (with bolts) (125 Only)	23,437	23,437	23,437			1			1
HI-TRAC 125 Neutron Shield Jacket Water	8,281	8,281	8,281		1	1			1
HI-TRAC 125 D Neutron Shield Jacket Water	9,000	9,000	9,000		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	230	480			1	1		1

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

<sup>†</sup> See Table 8.1.2 for a description of each load handling case.

<sup>††</sup> Add an additional 1955 lbs. for the HI-STORM 100A overpack.

TABLE 8.1.2  
ESTIMATED HANDLING WEIGHTS  
125-TON HI-TRAC\*\*

**Caution:**

The maximum weight supported by the 125-Ton HI-TRAC lifting trunnions cannot exceed 250,000 lbs. Users must take actions to ensure that this limit is not exceeded.

**Note:**

The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine their specific handling weights based on the MPC contents and the expected handling modes.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank empty)	231,300	239,300	237,800
2	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank full)	239,500	247,600	246,100
3	Loaded HI-TRAC 125 During Movement through Hatchway	236,500	244,300	243,700
1A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank empty)	232,800	240,900	239,400
2A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank full)	241,800	249,900	248,400
3A	Loaded HI-TRAC 125D During Movement through Hatchway	227,300	235,100	234,500
4	MPC during transfer operations	80,467	88,315	87,721
5A	Loaded HI-STORM 100 in storage (See Note 5 to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (232) in storage (See Note 5 to Table 8.1.1)	335,500	343,600	342,800
5C	Loaded HI-STORM 100S (243) in storage (See Note 5 to Table 8.1.1)	344,500	352,600	351,800
6	Loaded HI-TRAC and transfer frame during on site handling	239,434	247,282	246,688

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.3  
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS 100-  
TON HI-TRAC\*\*

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case <sup>†</sup> Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid) <sup>††</sup>	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (232) overpack (without lid) <sup>††</sup>	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid) <sup>††</sup>	239,000	239,000	239,000					1	
HI-STORM 100S lid (without rigging)	25,500	25,500	25,500						
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,200	3,200	3,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Empty HI-TRAC (without Top Lid, neutron shield jacket water, or bottom lids)	84,003	84,003	84,003	1	1	1			1
HI-TRAC Top Lid	1,189	1,189	1,189			1			1
HI-TRAC Pool Lid	7,863	7,863	7,863	1	1				
HI-TRAC Transfer Lid	16,686	16,686	16,686			1			1
HI-TRAC Neutron Shield Jacket Water	7,583	7,583	7,583		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480				1		1

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

<sup>†</sup> See Table 8.1.4 for a description of each load handling case.

<sup>† †</sup> Add an additional 1955 lbs. for the HI-STORM 100A overpack.

Table 8.1.4  
ESTIMATED HANDLING WEIGHTS  
100-TON HI-TRAC\*\*

**Caution:**

The maximum weight supported by the 100-Ton HI-TRAC lifting trunnions cannot exceed 200,000 lbs. Users must take actions to ensure that this limit is not exceeded.

**Note:**

The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly and therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine the handling weights based on the contents to be loaded and the expected mode of operations.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC removal from spent fuel pool (neutron tank empty)	192,844	200,942	199,425
2	Loaded HI-TRAC removal from spent fuel pool (neutron tank full)	200,427	208,525	207,008
3	Loaded HI-TRAC During Movement through Hatchway	192,647	200,745	199,901
4	MPC during transfer operations	80,467	88,565	87,721
5A	Loaded HI-STORM 100 in storage (See Note 5 to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (232) in storage (See Note 5 to Table 8.1.1)	335,500	343,600	342,700
5C	Loaded HI-STORM 100S (243) in storage (See Note 5 to Table 8.1.1)	344,500	352,600	351,700
6	Loaded HI-TRAC and transfer frame during on site handling	196,627	204,725	203,881

\*\* Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.5  
HI-STORM 100 SYSTEM TORQUE REQUIREMENTS

Fastener <sup>†</sup>	Torque (ft-lbs) <sup>††</sup>	Pattern <sup>†††</sup>
HI-TRAC Top Lid Bolts <sup>†</sup>	Hand tight	None
HI-TRAC Pool Lid Bolts (36 Bolt Lid) <sup>†</sup>	58 ft-lbs	Figure 8.1.37
HI-TRAC Pool Lid Bolts (16 Bolt Lid) <sup>†</sup>	110 ft-lbs	Figure 8.1.37
100-Ton HI-TRAC Transfer Lid Bolts <sup>†</sup>	203 ft-lbs	Figure 8.1.37
125-Ton HI-TRAC Transfer Lid Bolts <sup>†</sup>	270 ft-lbs	Figure 8.1.37
MPC Lift Cleats Stud Nuts <sup>†</sup>	793 ft-lbs	None
MPC Lift Hole Plugs <sup>†</sup>	Hand tight	None
Threaded Fuel Spacers	Hand Tight	None
HI-STORM Lid Nuts <sup>†</sup>	100 ft-lbs	None
HI-STORM 100S Lid Nuts <sup>†</sup> (Temporary and Permanent Lids)	Hand Tight +1/8 to 1/4 turn	None
Door Locking Pins	Hand Tight + 1/8 to 1/2 turn	None
HI-STORM 100 Vent Screen/Temperature Element Screws	Hand Tight	None
HI-STORM 100A Anchor Studs	55- 65 ksi tension applied by bolt tensioner (no initial torque)	None

<sup>†</sup> Studs and nuts shall be cleaned and inspected for damage or excessive thread wear (replace if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

<sup>††</sup> Unless specifically specified, torques have a +/- 5% tolerance.

<sup>†††</sup> No detorquing pattern is needed.

**Table 8.1.6**  
**HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION**

<b>Equipment</b>	<b>Important To Safety Classification</b>	<b>Reference Figure<sup>†</sup></b>	<b>Description</b>
Air Pads/Rollers	Not Important To Safety	8.1.29	Used for HI-STORM or HI-TRAC cask positioning. May be used in conjunction with the cask transporter or other HI-STORM 100 or HI-TRAC lifting device.
Annulus Overpressure System	Not Important To Safety	8.1.14	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Annulus Shield	Not Important To Safety	8.1.13	A shield that is placed at the top of the HI-TRAC annulus to provide supplemental shielding to the operators performing cask loading and closure operations.
Automated Welding System	Not Important To Safety	8.1.2b	Used for remote field welding of the MPC.
AWS Baseplate Shield	Not Important To Safety	8.1.2b	Provides supplemental shielding to the operators during the cask closure operations.
Bottom Lid Transfer Slide (Not used with HI-TRAC 125D)	Not Important To Safety	8.1.26	Used to simultaneously replace the pool lid with the transfer lid under the suspended HI-TRAC and MPC. Used in conjunction with the bottom lid transfer step.
Cask Transporter	Not Important to Safety unless site-specific conditions require transfer cask or overpack handling outside drop analysis basis.	8.1.29a and 8.1.29b	Used for handling of the HI-STORM 100 Overpack and/or the HI-TRAC Transfer Cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function.
Cool-Down System	Not Important To Safety	8.3.4	A closed-loop forced ventilation cooling system used to gas-cool the MPC fuel assemblies down to a temperature at which water can be introduced without the risk of uncontrolled pressure transients in the MPC due to flashing or thermally shocking the fuel assemblies. The cool-down system is attached between the MPC drain and vent ports. The cool-down system is used only for unloading operations.

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<sup>†</sup> Figures are representative and may not depict all configurations for all users.

Table 8.1.6  
**HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION**  
(Continued)

<b>Equipment</b>	<b>Important To Safety Classification</b>	<b>Reference Figure<sup>†</sup></b>	<b>Description</b>
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	8.1.9	Used for rigging such components such as the HI-TRAC top lid, pool lid, MPC lid, transfer lid, AWS, HI-STORM Lid and auxiliary shielding and the empty MPC.
Helium Backfill System	Not Important To Safety	8.1.23	Used for controlled insertion of helium into the MPC for leakage testing, blowdown and placement into storage.
HI-STORM 100 Lifting Jacks	Not Important To Safety	8.1.36	Jack system used for lifting the HI-STORM overpack to provide clearance for inserting or removing a device for transportation. .
Alignment Device	Not Important To Safety	8.1.31	Guides HI-TRAC into place on top of HI-STORM for MPC transfers. (Not used for HI-TRAC 125D)
HI-STORM Lifting Devices	Determined site-specifically based on type, location, and height of lift being performed. Lifting devices shall be provided in accordance with ANSI N14.6.	Not shown.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM 100 for cask handling. Does not include the crane hook (or other primary lifting device) device.
HI-STORM Vent Duct Shield Inserts	Important to Safety Category C .	8.1.30	Used for prevention of radiation streaming from the HI-STORM 100 exit vents during MPC transfers to and from HI-STORM. Not used with the HI-STORM 100S.
HI-TRAC Lid Spacer	Spacer Ring is Not-Important-To-Safety, Studs or bolts are I Important to Safety Category B	Not Shown	Optional ancillary which is used during MPC transfer operations to increase the clearance between the top of the MPC and the underside of the HI-TRAC top lid. Longer threaded studs (or bolts), supplied with the lid spacer, replace the standard threaded studs (or bolts) supplied with the HI-TRAC. The HI-TRAC lid spacer may ONLY be used when the HI-TRAC is handled in the vertical orientation or if HI-TRAC transfer lid is NOT used. The height of the spacer shall be limited to ensure that the weights and C.G. heights in a loaded HI-TRAC with the spacer do not exceed the bounding values found in Section 3.2 of the FSAR.
HI-TRAC Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC handling shall be provided in accordance with ANSI N14.6.	8.1.3	Used for connecting the crane (or other primary lifting device) to the HI-TRAC for cask handling. Does not include the crane hook (or other primary lifting device).

<sup>†</sup> Figures are representative and may not depict all configurations for all users.



**Table 8.1.6**  
**HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION**  
(Continued)

<b>Equipment</b>	<b>Important To Safety Classification</b>	<b>Reference Figure<sup>†</sup></b>	<b>Description</b>
HI-TRAC transfer frame	Not Important To Safety	8.1.4	A steel frame used to support HI-TRAC during delivery, on-site movement and upending/downending operations.
Cask Primary Lifting Device (Cask Transfer Facility)	Important to Safety. Quality classification of subcomponents determined site-specifically.	8.1.28 and 8.1.32	Optional auxiliary (Non-Part 50) cask lifting device(s) used for cask upending and downending and HI-TRAC raising for positioning on top of HI-STORM to allow MPC transfer. The device may consist of a crane, lifting platform, gantry system or any other suitable device used for such purpose.
Inflatable Annulus Seal	Not Important To Safety	8.1.13	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System	Important to Safety Status determined by each licensee. MPC lid lifting portions of the Lid Retention System shall meet the requirements of ANSI N14.6.	8.1.15, 8.1.17	Optional. The Lid Retention System secures the MPC lid in place during cask handling operations between the pool and decontamination pad.
MPC Lift Cleats	Important To Safety – Category A. MPC Lift Cleats shall be provided in accordance with of ANSI N14.6.	8.1.24	MPC lift cleats consist of the cleats and attachment hardware. The cleats are supplied as solid steel components that contain no welds. The MPC lift cleats are used to secure the MPC inside HI-TRAC during bottom lid replacement and support the MPC during MPC transfer from HI-TRAC into HI-STORM and vice versa. The ITS classification of the lifting device attached to the cleats may be lower than the cleat itself, as determined site-specifically.
Hydrostatic Test System	Not Important to Safety	8.1.20	Used to pressure test the MPC lid-to-shell weld.
MPC Downloader	Important To Safety status determined site-specifically. MPC Downloader Shall meet the requirements of CoC, Appendix B, Section 3.5 .	8.1.28 and 8.1.32	A lifting device used to help raise and lower the MPC during MPC transfer operations to limit the lift force of the MPC against the top lid of HI-TRAC. The MPC downloader may take several forms depending on the location of MPC transfer and may be used in conjunction with other lifting devices.

<sup>†</sup> Figures are representative and may not depict all configurations for all users.

Table 8.1.6  
 HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION  
 (Continued)

Equipment	Important To Safety Classification	Reference Figure <sup>†</sup>	Description
Deleted			
Deleted			
Mating Device	Important-To-Safety – Category B	8.1.31	Used to mate HI-TRAC 125D to HI-STORM during transfer operations. Includes sliding drawer for use in removing HI-TRAC pool lid.
MPC Support Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	8.1.25	Used to secure the MPC to the lift yoke during HI-TRAC bottom lid replacement operations. Attaches between the MPC lift cleats and the lift yoke. Can be configured for different crane hook configuration.
MPC Upending Frame	Not Important to Safety	8.1.6	A steel frame used to evenly support the MPC during upending operations. and control the upending process.
<i>Supplemental Cooling System (MSLD) (Helium Leakage Detector)</i>	<del>Not Important To Safety</del> <i>Not Important to Safety</i>	<del>2.C.1</del> <i>Not shown</i>	<i>A system used to circulate water or other coolant through the HI-TRAC annulus in order to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits during operations with the MPC in the HI-TRAC. Required only for MPC containing high burn-up fuel as determined in accordance with Section 4.5. Used for helium leakage testing of the MPC closure welds.</i>
Deleted			
Deleted			
Temporary Shield Ring	Not Important To Safety	8.1.18	A water-filled tank that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.
Vacuum Drying (Moisture Removal) System	Not Important To Safety	8.1.22a	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Not Important To Safety	8.1.22b	Used for removal of residual moisture from the MPC following water draining.
Vent and Drain RVOAs	Not Important To Safety	8.1.16	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Deleted			
Weld Removal System	Not Important To Safety	8.3.2b	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

<sup>†</sup> Figures are representative and may not depict all configurations for all users.

Table 8.1.7  
 HI-STORM 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND  
 UNLOADING OPERATIONS<sup>†</sup>

<b>Instrument</b>	<b>Function</b>
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors fluid flow rate during various loading and unloading operations.
Deleted	
<del>Helium Mass Spectrometer Leak Detector (MSLD)</del> <i>Deleted</i>	<del>Ensures leakage rates of welds are within acceptance criteria.</del>
Deleted	
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Deleted	
Temperature Surface Pyrometer	For HI-STORM vent operability testing.
Vacuum Gages	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Deleted	
Deleted	
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

<sup>†</sup> All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 8.1.8  
HI-STORM 100 SYSTEM OVERPACK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100 overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100 Overpack Lid:

1. Lid studs and nuts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device/ holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.

HI-STORM 100 Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be available, intact, and free of holes and tears in the fabric.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.
7. Anchor hardware, if used, shall be checked for general condition.

Table 8.1.9  
MPC INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:

1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Vent and Drain attachments shall be inspected for availability, thread condition operability and general condition.
4. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
5. Lower fuel spacers (if used) shall be inspected for availability and general condition.
6. Drain and vent port cover plates shall be inspected for availability and general condition.
7. Serial numbers shall be inspected for readability.

MPC Main Body:

1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.

Table 8.1.10  
HI-TRAC TRANSFER CASK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-TRAC Top Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All Top Lid surfaces shall be relatively free of dents, scratches, gouges or other damage.

HI-TRAC Main Body:

1. The painted surfaces shall be inspected for corrosion, chipped, cracked or blistered paint.
2. The Top Lid bolt holes shall be inspected for dirt, debris and thread damage.
3. The Top Lid lift holes shall be inspected for thread condition.
4. Lifting trunnions shall be inspected for deformation, cracks, end plate damage, corrosion, excessive galling, damage to the locking plate, presence or availability of locking plate and end plate retention bolts.
5. Pocket trunnion, if used, recesses shall be inspected for indications of overstressing (i.e., cracks, deformation, and excessive wear).
6. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
7. The nameplate shall be inspected for presence and general condition.
8. The neutron shield jacket shall be inspected for leaks.
9. Neutron shield jacket pressure relief valve shall be inspected for presence, and general condition.
10. The neutron shield jacket fill and drain plugs shall be inspected for presence, leaks, and general condition.
11. Bottom lid flange surface shall be clean and free of large scratches and gouges.

Table 8.1.10 (Continued)  
HI-TRAC OVERPACK INSPECTION CHECKLIST

HI-TRAC Transfer Lid (Not used with HI-TRAC 125D):

1. The doors shall be inspected for smooth actuation.
2. The threads shall be inspected for general condition.
3. The bolts shall be inspected for indications of overstressing (i.e., cracks, deformation, thread damage, excessive wear) and replaced as necessary.
4. Door locking pins shall be inspected for indications of overstressing (i.e., cracks, and deformation, thread damage, excessive wear) and replaced as necessary.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Lifting holes shall be inspected for thread damage.

HI-TRAC Pool Lid:

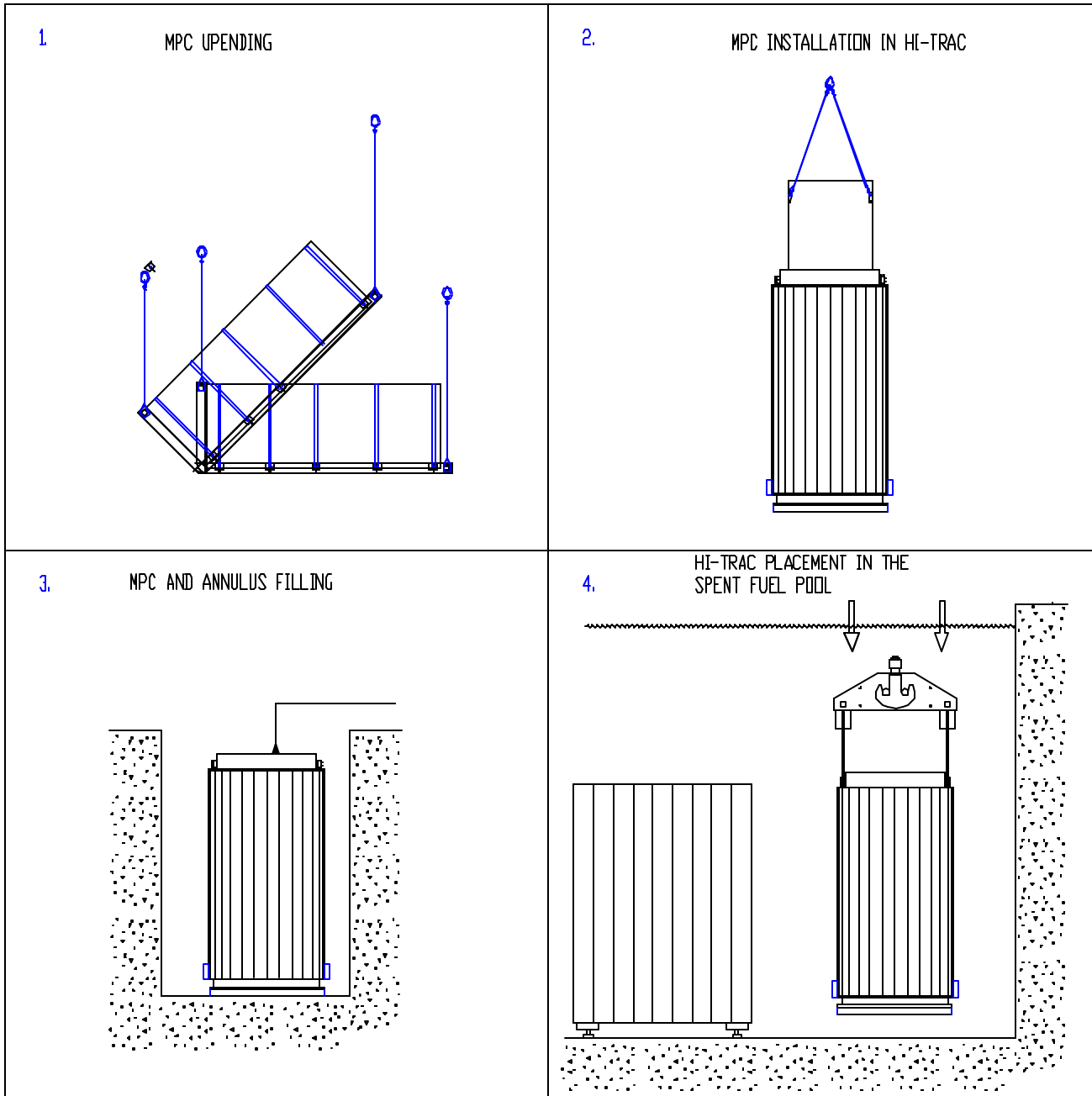
1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Threads shall be inspected for indications of damage.

<b>LOCATION: CASK RECEIVING AREA</b>
REMOVE HI-TRAC TOP LID
CONFIGURE HI-TRAC WITH POOL LID
INSTALL MPC IN HI-TRAC
INSTALL UPPER FUEL SPACERS
INSTALL LOWER FUEL SPACERS
FILL MPC AND ANNULUS
INSTALL ANNULUS SEAL
PLACE HI-TRAC IN SPENT FUEL POOL
<b>LOCATION: SPENT FUEL POOL</b>
LOAD FUEL ASSEMBLIES INTO MPC
PERFORM ASSEMBLY IDENTIFICATION VERIFICATION
INSTALL DRAIN LINE TO MPC LID
ALIGN MPC LID AND LIFT YOKE TO DRAIN LINE
INSTALL MPC LID
REMOVE HI-TRAC FROM SPENT FUEL POOL AND PLACE IN PREPARATION AREA
<b>LOCATION: CASK PREPARATION AREA</b>
DECONTAMINATE HI-TRAC BOTTOM
SET HI-TRAC IN CASK PREPARATION AREA
FILL NEUTRON WATER JACKET
MEASURE DOSE RATES AT MPC LID
DECONTAMINATE HI-TRAC AND LIFT YOKE
INSTALL TEMPORARY SHIELD RING
REMOVE INFLATABLE ANNULUS SEAL
LOWER ANNULUS WATER LEVEL SLIGHTLY
SMEAR MPC LID TOP SURFACES
INSTALL ANNULUS SHIELD
LOWER MPC WATER LEVEL
WELD MPC LID & PERFORM NDE
DELETED
RAISE MPC WATER LEVEL
PERFORM HYDRO TEST ON MPC
PERFORM LEAKAGE TESTING
DRAIN MPC
MEASURE VOLUME OF WATER DRAINED
DRY MPC
PERFORM MPC DRYNESS VERIFICATION TEST

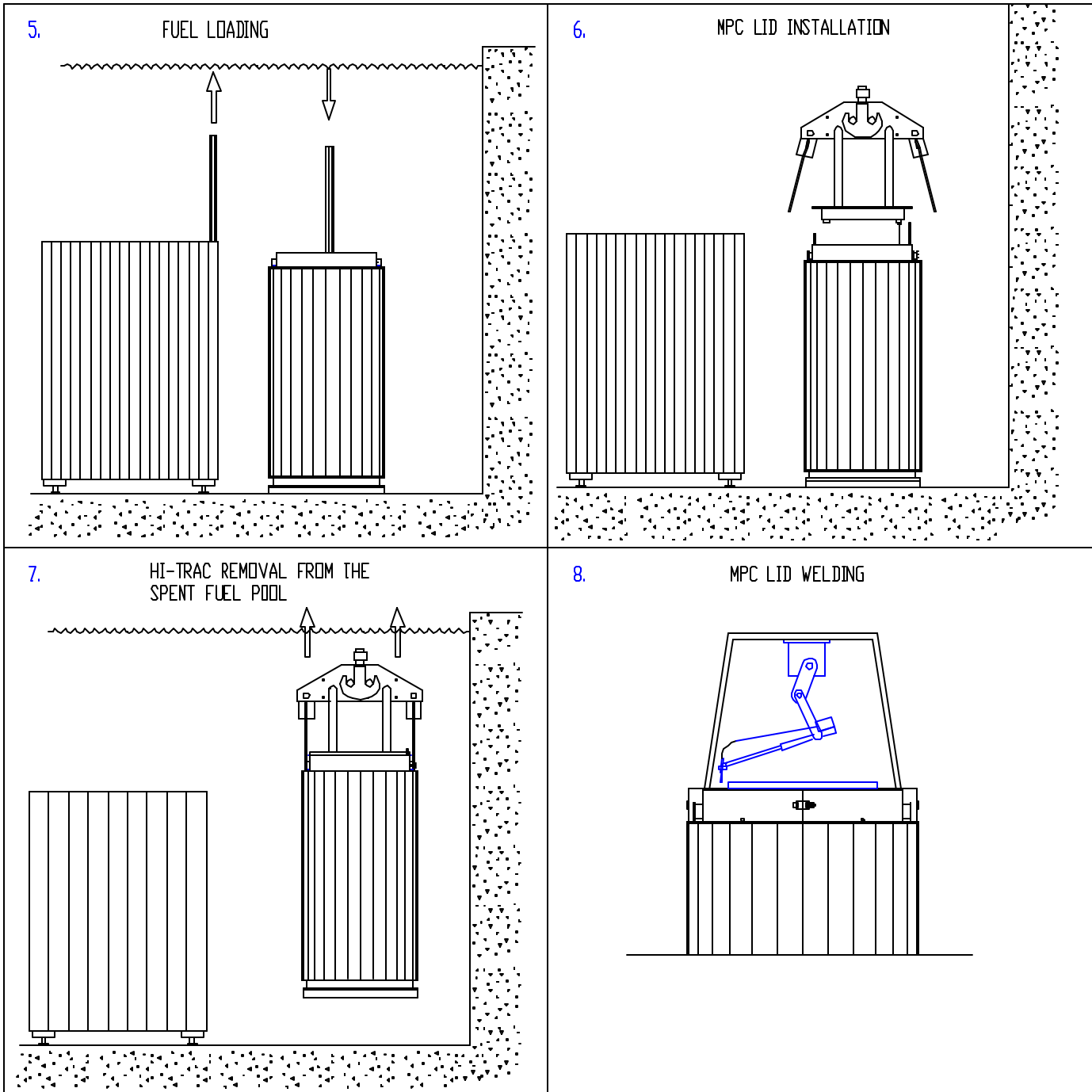
BACKFILL MPC
WELD VENT AND DRAIN PORT COVER PLATES & PERFORM NDE
WELD MPC CLOSURE RING & PERFORM NDE
DRAIN ANNULUS
INSTALL HI-TRAC TOP LID
PERFORM SURVEYS ON HI-TRAC
REMOVE TEMPORARY SHIELD RING
REPLACE POOL LID WITH TRANSFER LID (Not Required for HI-TRAC 125D)
INSTALL MPC LIFT CLEAT
PERFORM SURVEYS OF POOL LID
<b>LOCATION: CASK RECEIVING AREA</b>
POSITION HI-STORM FOR MPC TRANSFER
INSTALL ALIGNMENT DEVICE (MATING DEVICE FOR HI-TRAC 125D) & VENT DUCT SHIELD INSERTS
MATE OVERPACK AND TRANSFER CASK
ATTACH MPC SLINGS
OPEN TRANSFER LID DOORS (Remove pool lid and open mating device drawer for HI-TRAC 125D)
LOWER MPC INTO HI-STORM
REMOVE HI-TRAC FROM ON TOP OF HI-STORM
REMOVE MPC LIFT CLEAT
REMOVE ALIGNMENT DEVICE (Mating device for HI-TRAC 125D) & VENT DUCT SHIELD INSERTS
INSTALL HI-STORM LID
PERFORM SHIELDING EFFECTIVENESS TESTING
<b>LOCATION: ISFSI</b>
PLACE HI-STORM IN STORAGE
INSTALL GAMMA SHIELD CROSS PLATES AND THERMOCOUPLES IN HI-STORM 100 OVERPACK EXIT VENTS
INSTALL HI-STORM VENT SCREENS
PERFORM THERMAL TESTING

**Figure 8.1.1; Loading Operations Flow Diagram**

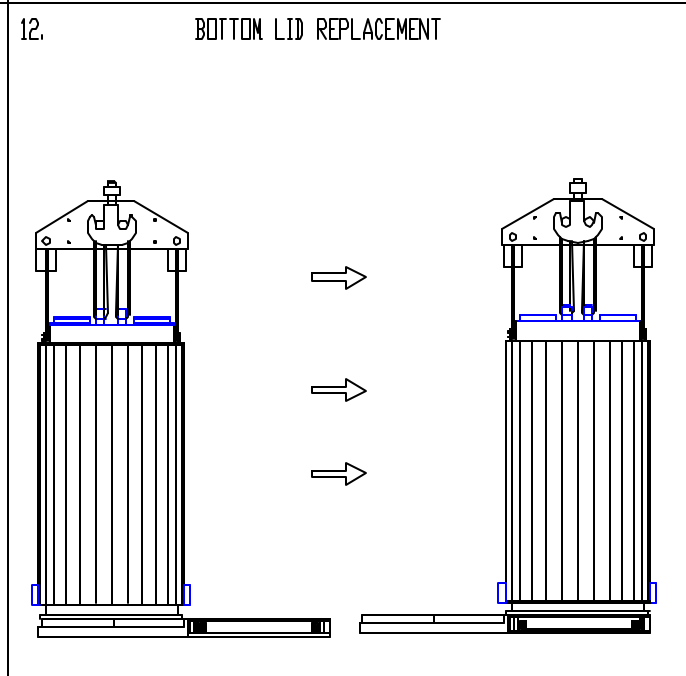
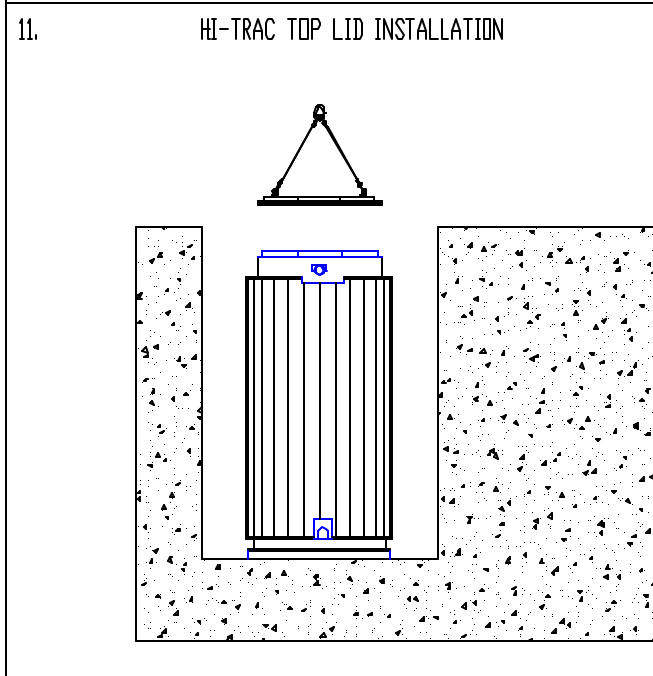
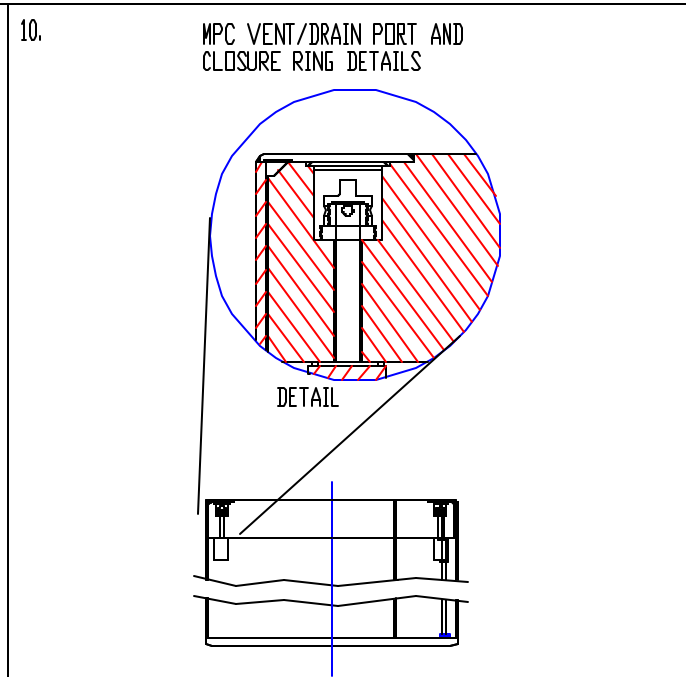
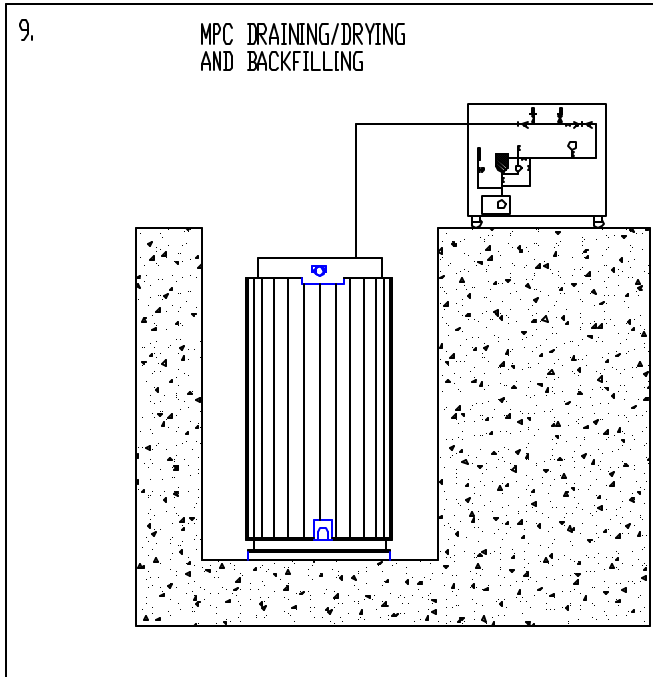




**Figure 8.1.2a; Major HI-STORM 100 Loading Operations**



**Figure 8.1.2b; Major HI-STORM 100 Loading Operations**

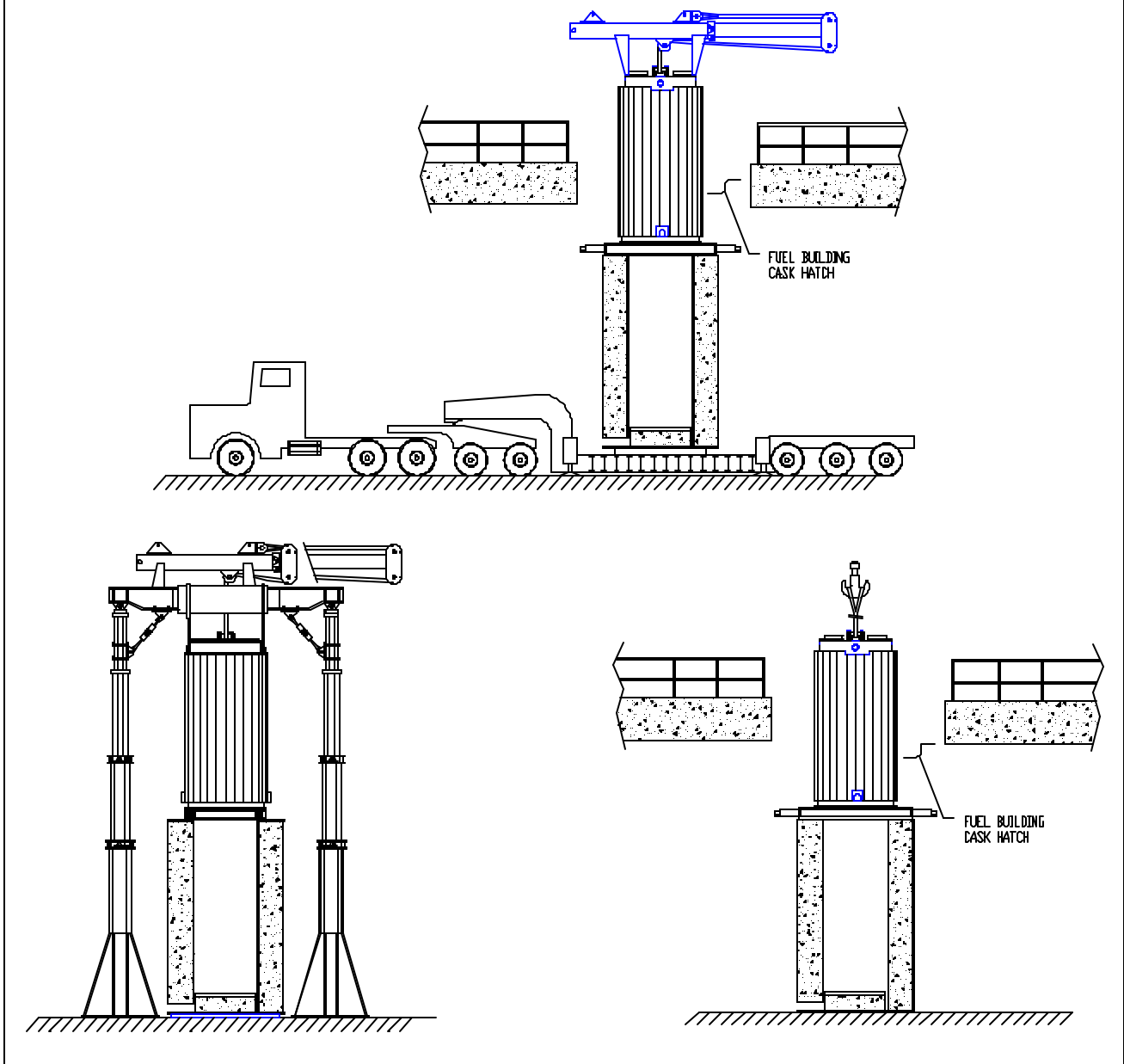


**Note: Bottom Lid Replacement is not required for HI-TRAC 125D**

**Figure 8.1.2c; Major HI-STORM 100 Loading Operations**

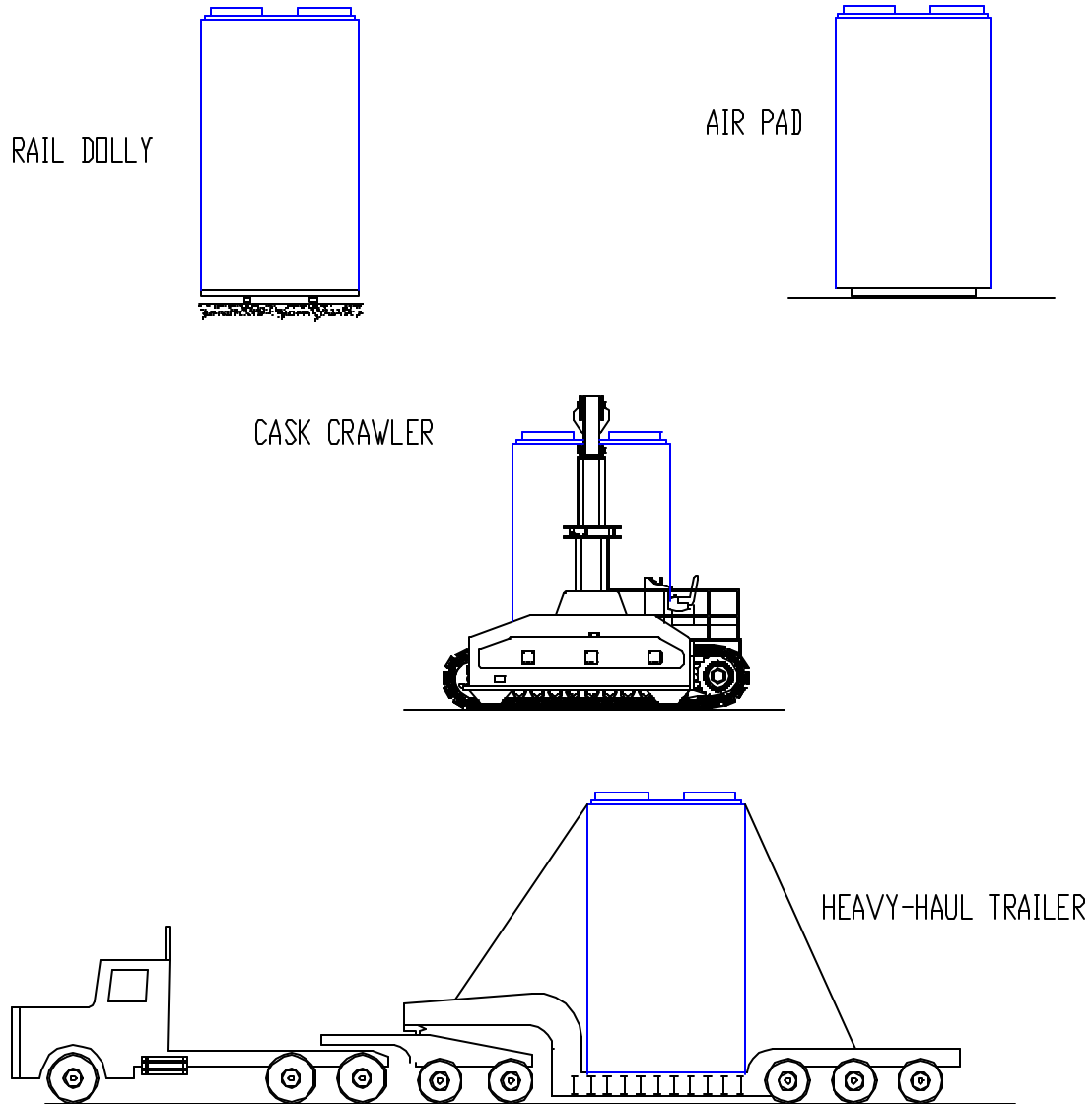
13.

SAMPLE MPC TRANSFER MODES



**Figure 8.1.2d; Major HI-STORM 100 Loading Operations(HI-TRAC with Transfer Lid Shown)**

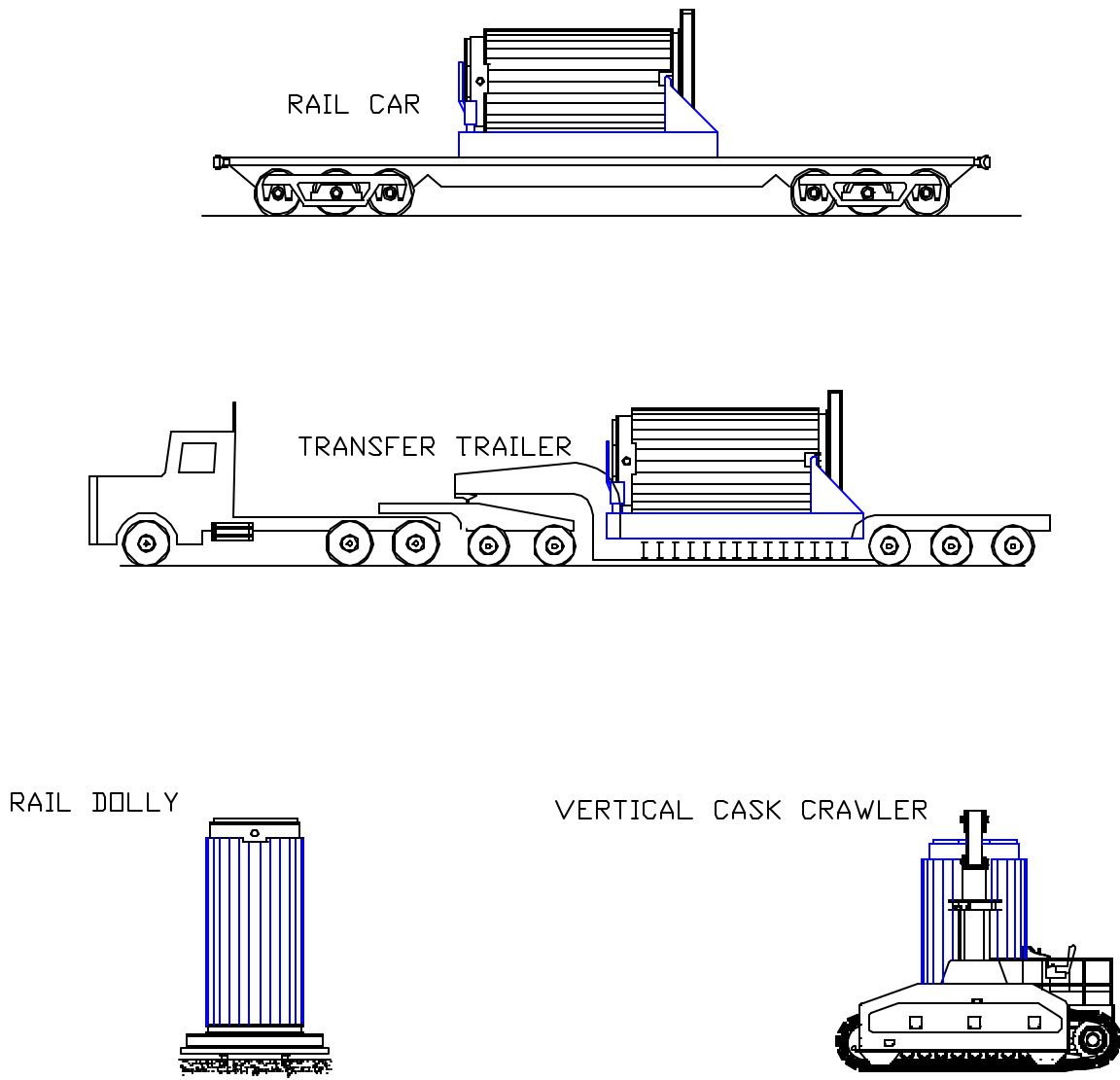
14. SAMPLE HI-STORM HANDLING METHODS



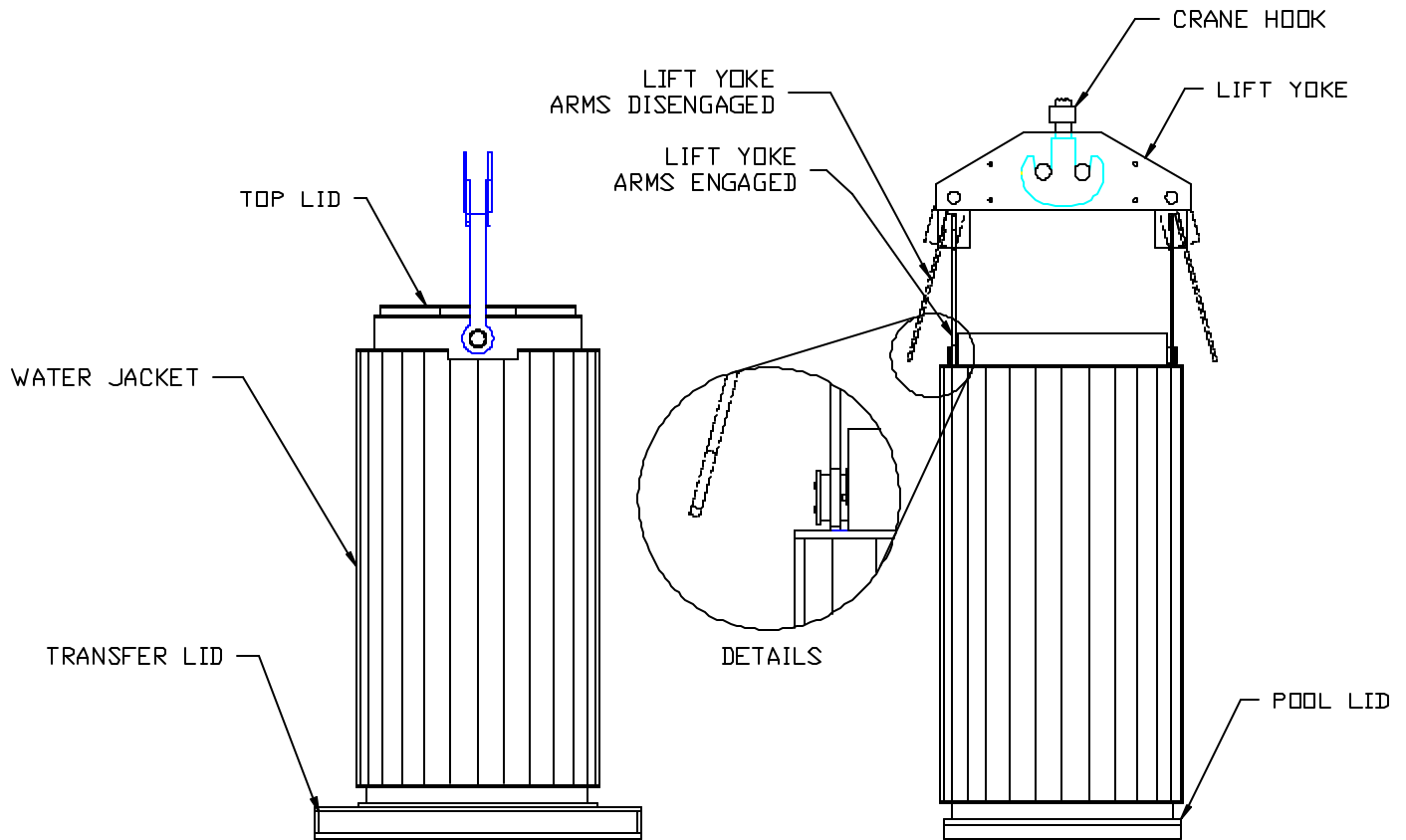
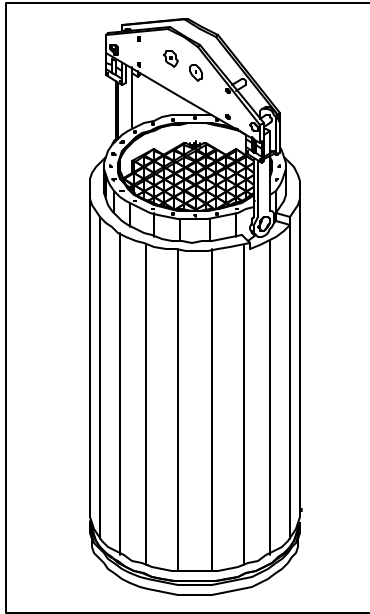
**Figure 8.1.2e; Example of HI-STORM 100 Handling Options**

15.

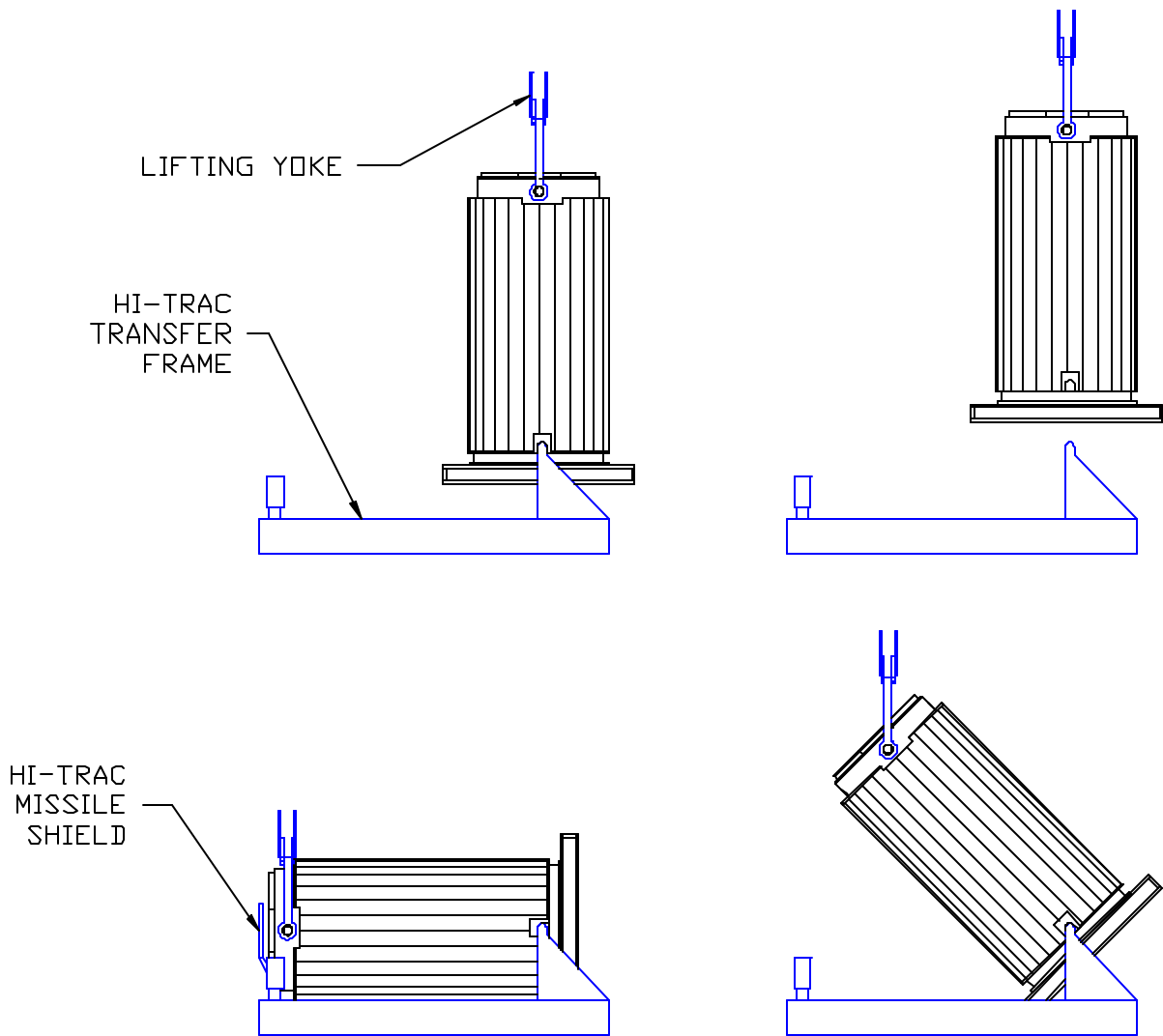
SAMPLE HI-TRAC HANDLING METHODS



**Figure 8.1.2f; Example of HI-TRAC Handling Options (Missile Shields Not Shown For Clarity)**



**Figure 8.1.3; Lift Yoke Engagement and Vertical HI-TRAC Handling (Shown with the Pool Lid and the Transfer Lid)**

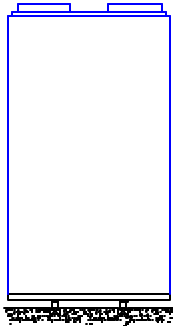


**Figure 8.1.4; HI-TRAC Upending/Downending in the Transfer Frame**

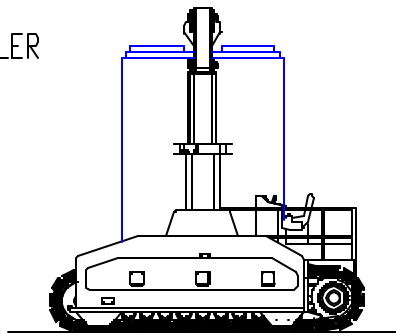
**(HI-TRAC with pocket trunnions shown, HI-TRAC 125D utilizes separate upending frame without pocket trunnions)**



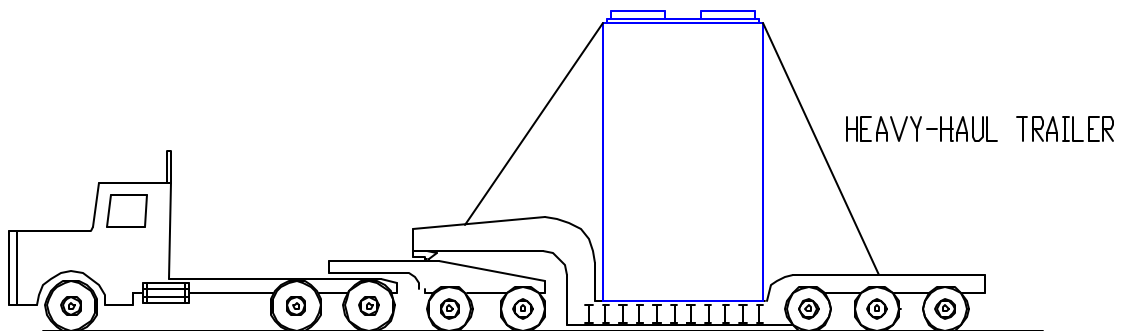
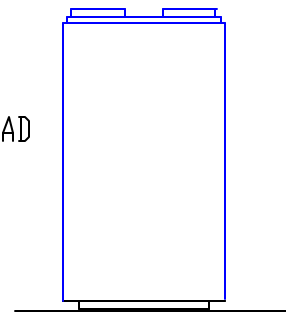
RAIL DOLLY



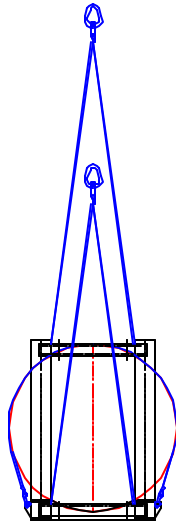
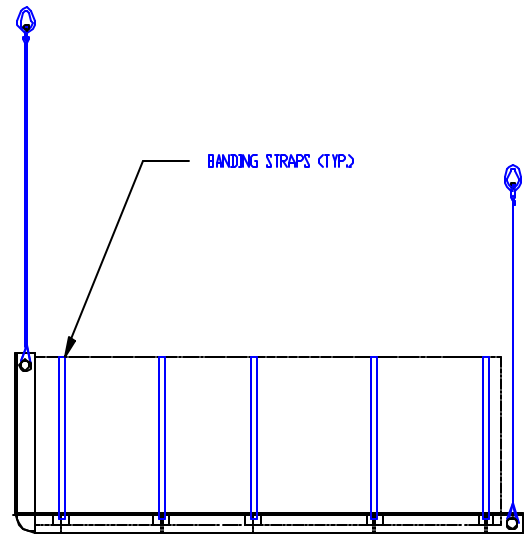
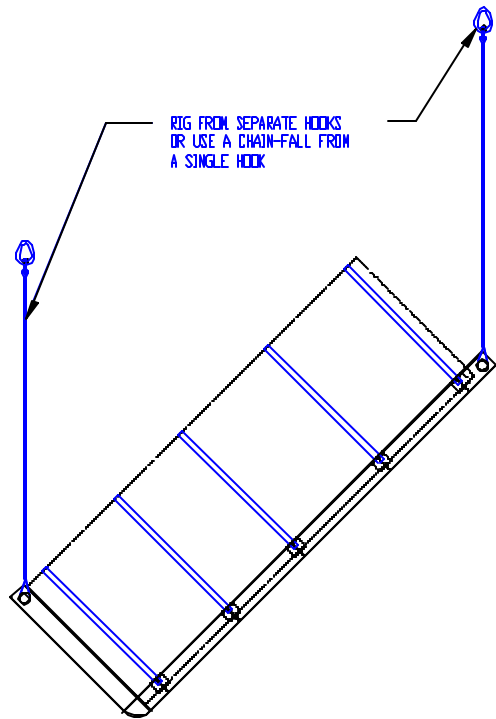
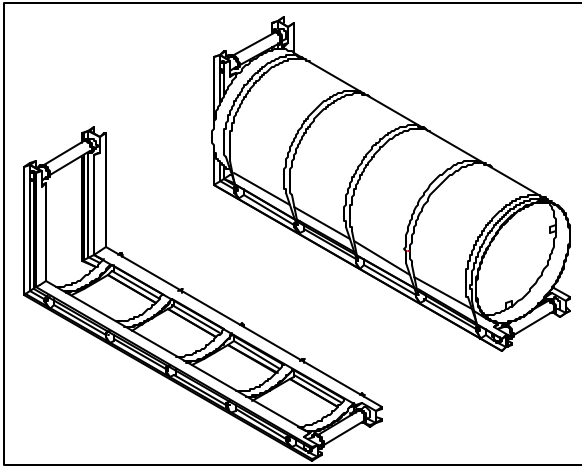
CASK CRAWLER



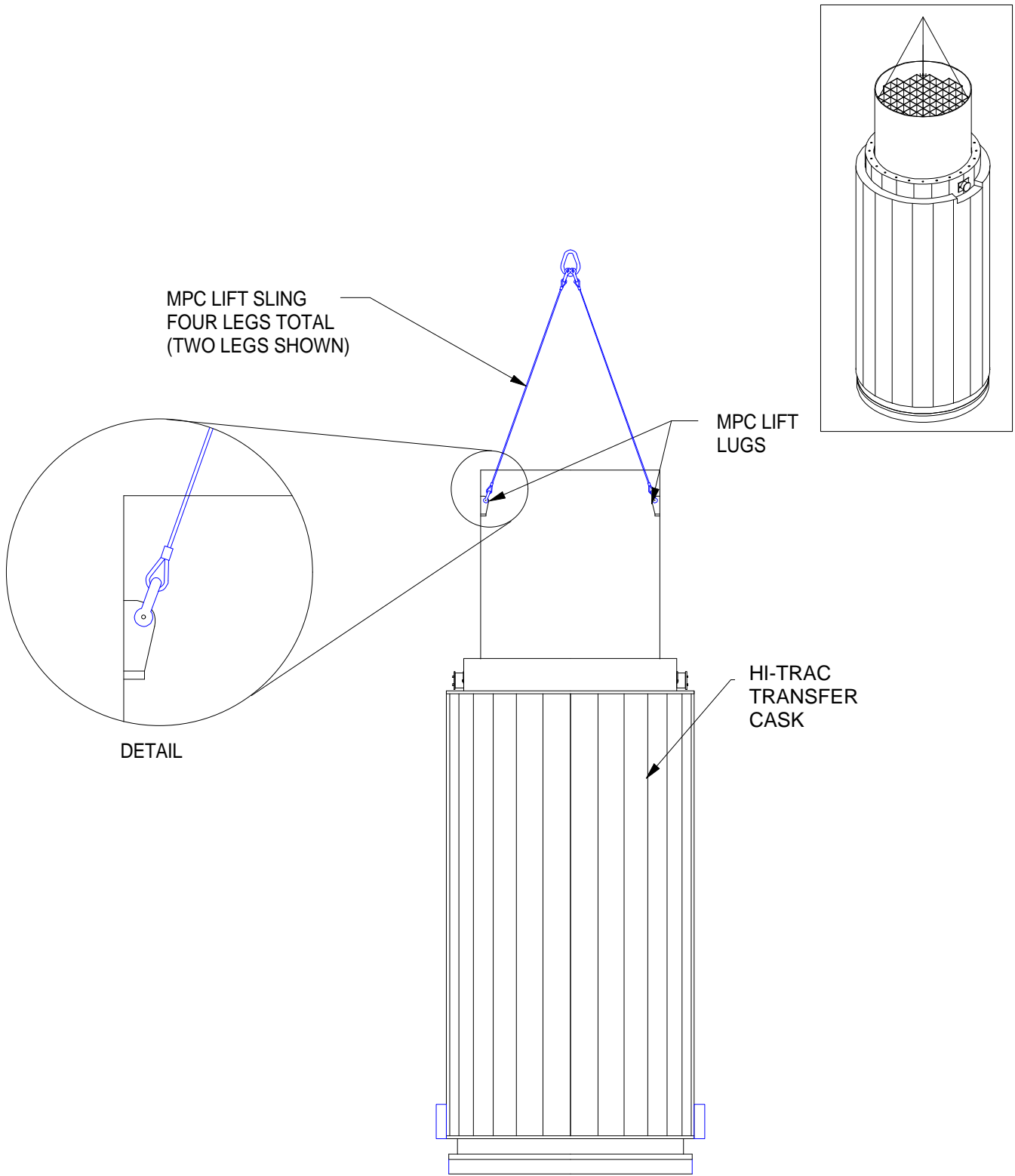
AIR PAD



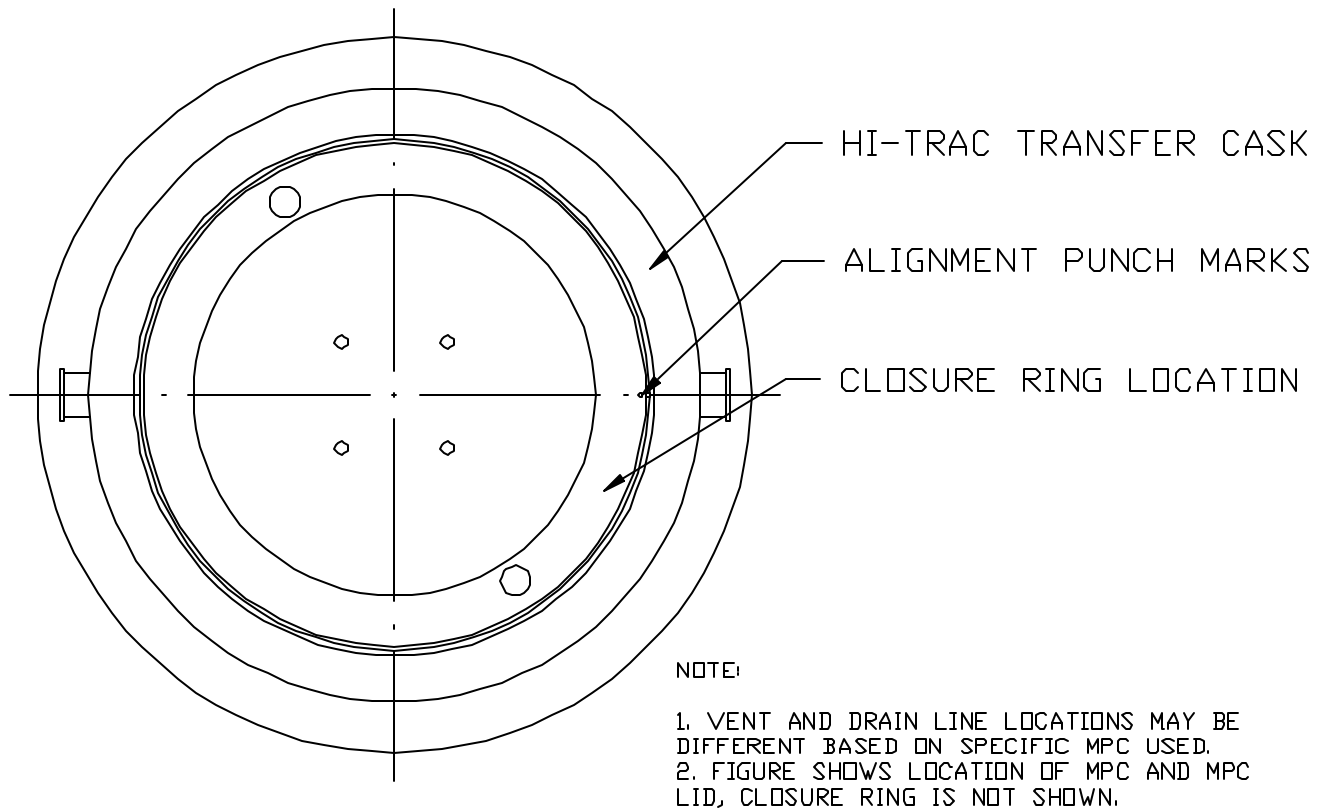
**Figure 8.1.5; HI-STORM Vertical Handling**



**Figure 8.1.6; MPC Upending in the MPC Upending Frame**

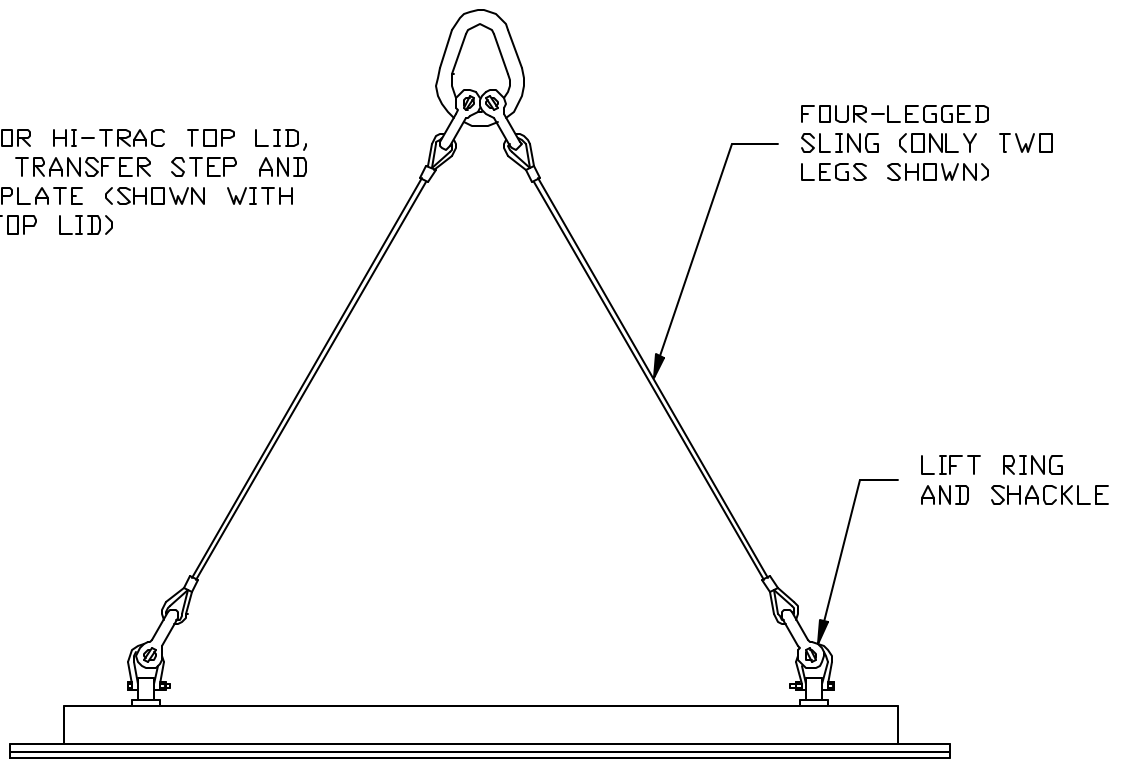


**Figure 8.1.7; MPC Rigging for Vertical Lifts**



**Figure 8.1.8; MPC Alignment in HI-TRAC**

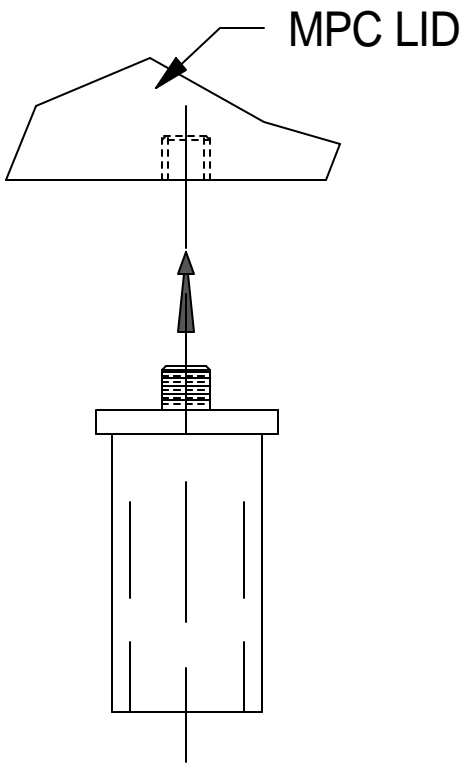
RIGGING FOR HI-TRAC TOP LID,  
POOL LID, TRANSFER STEP AND  
AWS BASEPLATE (SHOWN WITH  
HI-TRAC TOP LID)



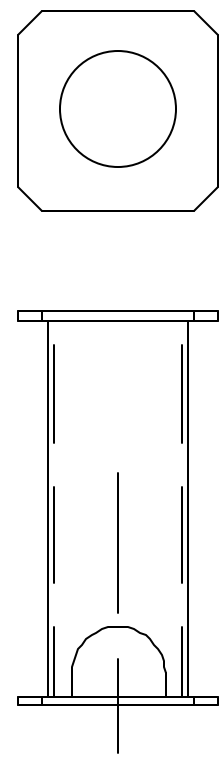
FOUR-LEGGED  
SLING (ONLY TWO  
LEGS SHOWN)

LIFT RING  
AND SHACKLE

**Figure 8.1.9; MPC Lid AND HI-TRAC Accessory Rigging**



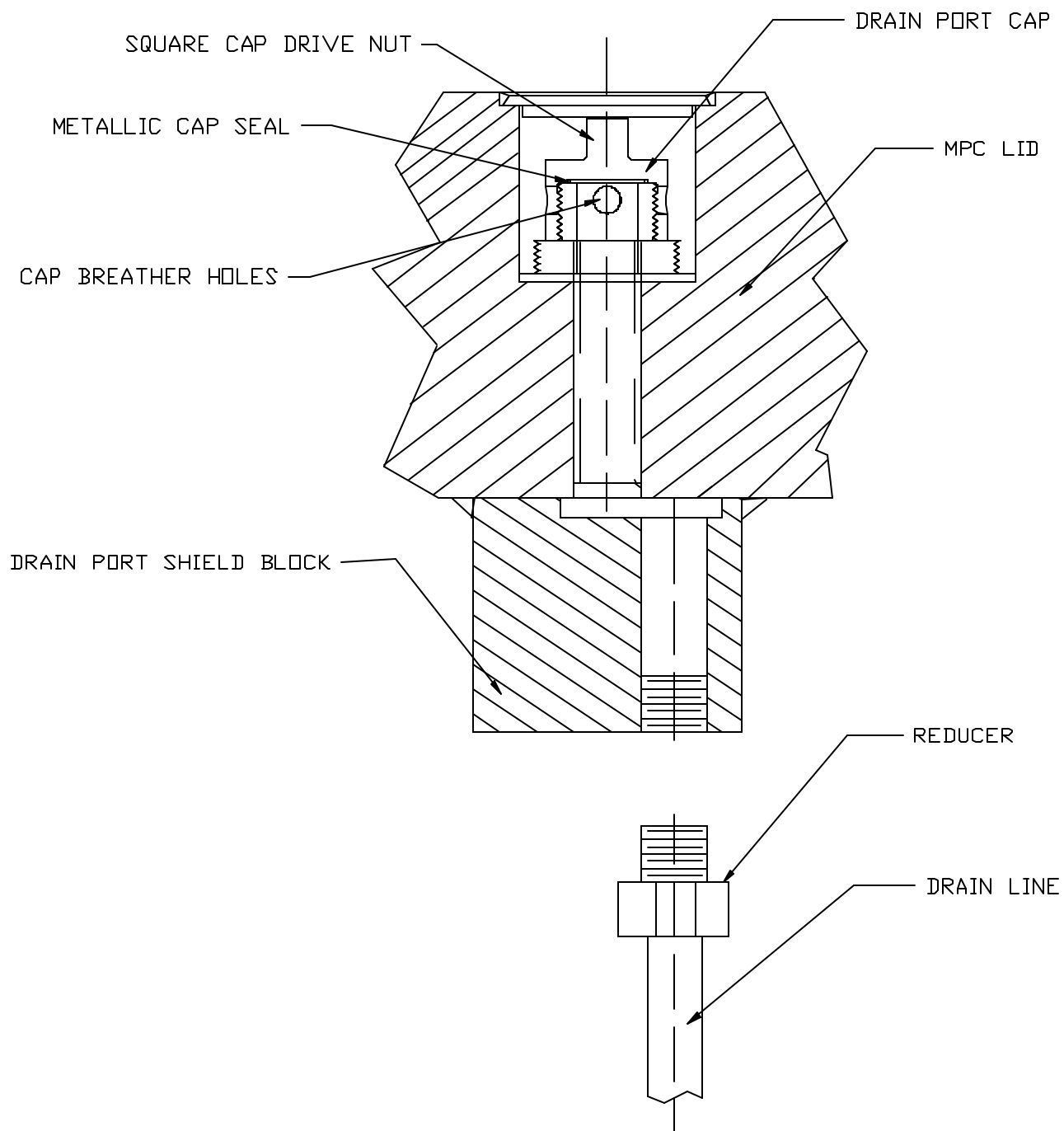
UPPER FUEL SPACER



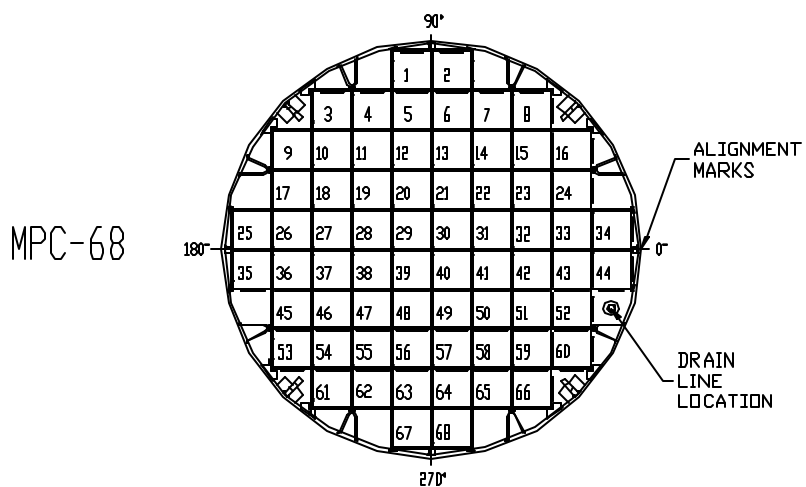
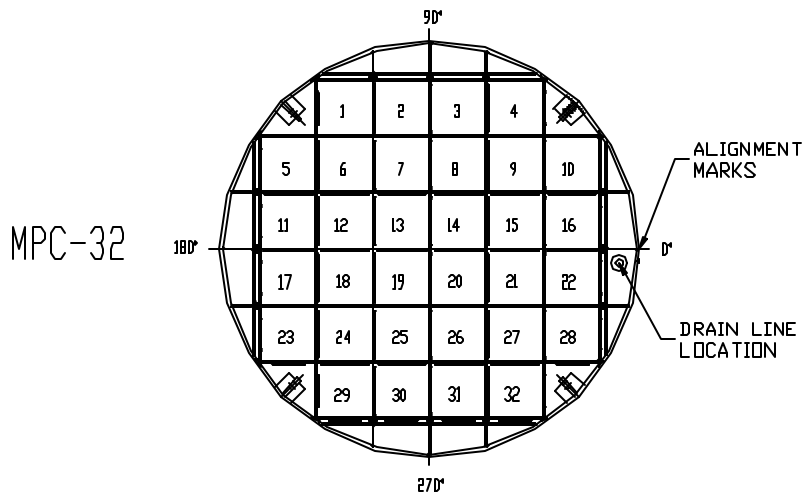
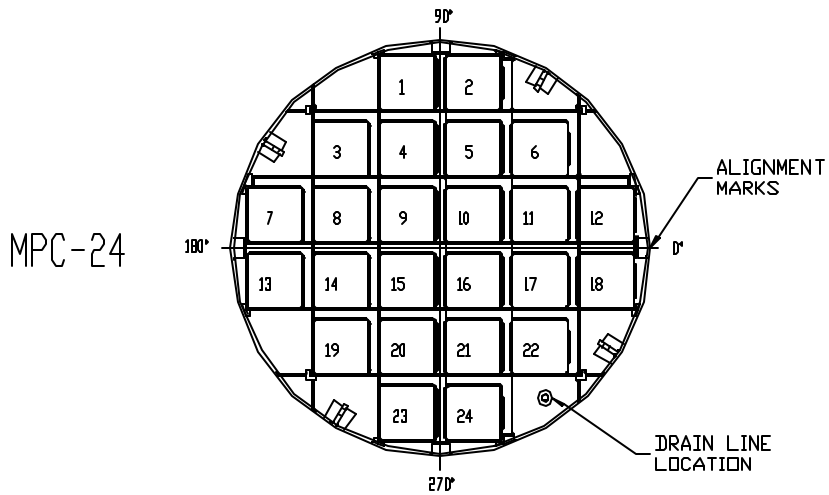
LOWER FUEL SPACER

Note: Lengths are based on specific fuel assembly type to be stored.

**Figure 8.1.10; Fuel Spacers**

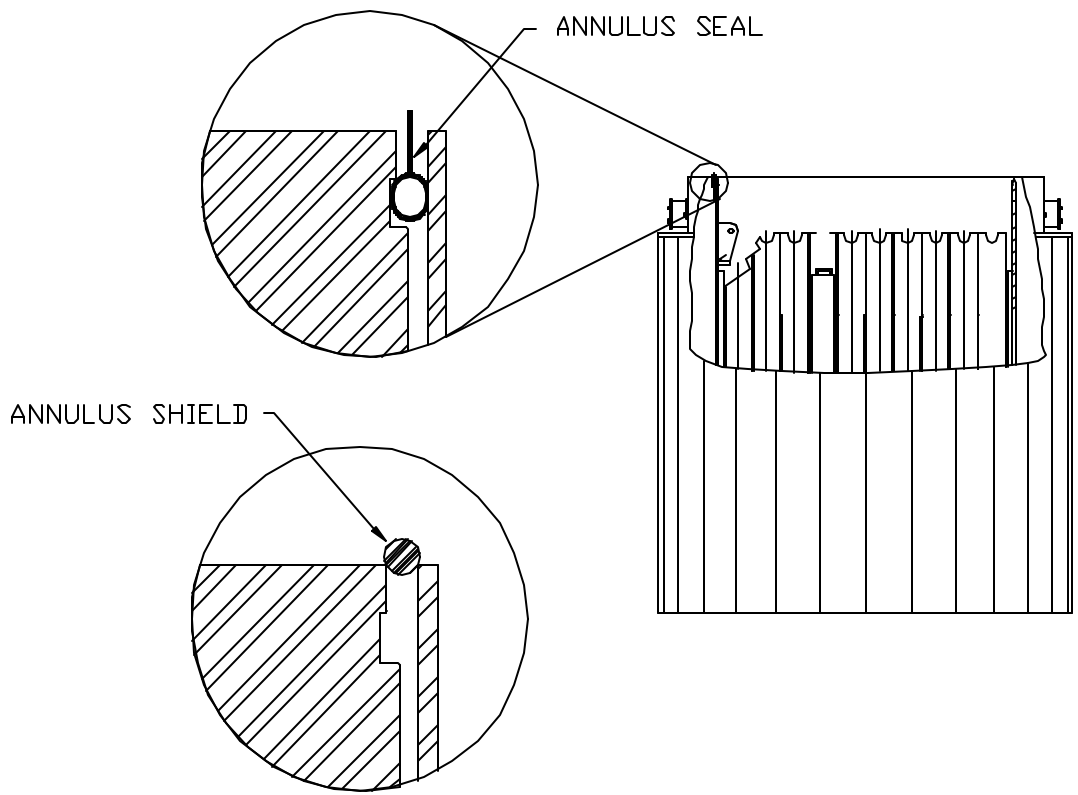


**Figure 8.1.11; Drain Port Details**



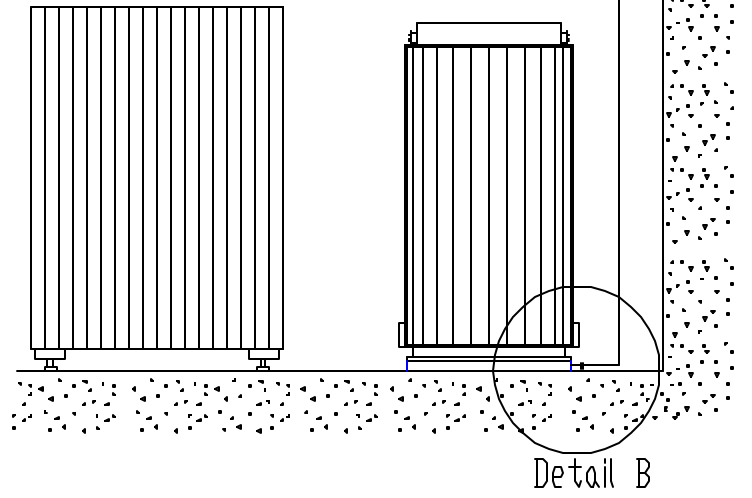
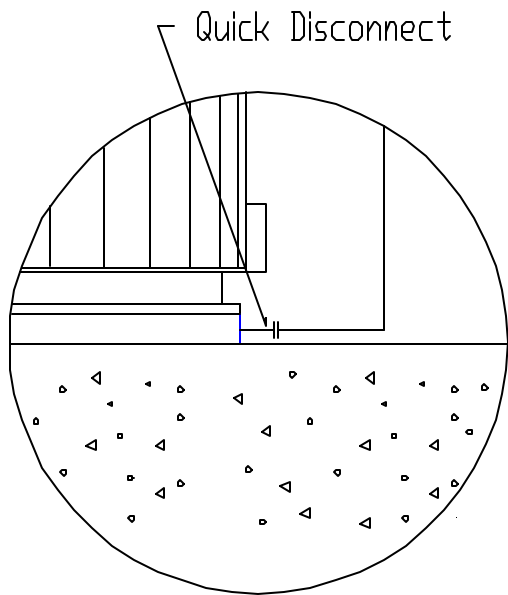
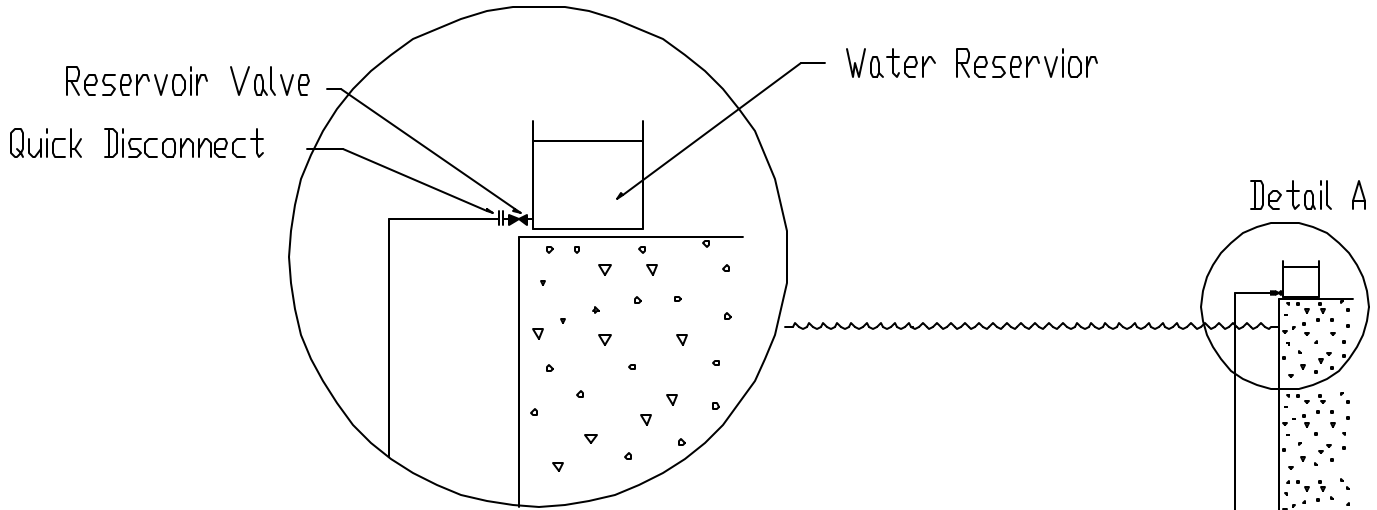
**Figure 8.1.12; Drain Line Positioning**





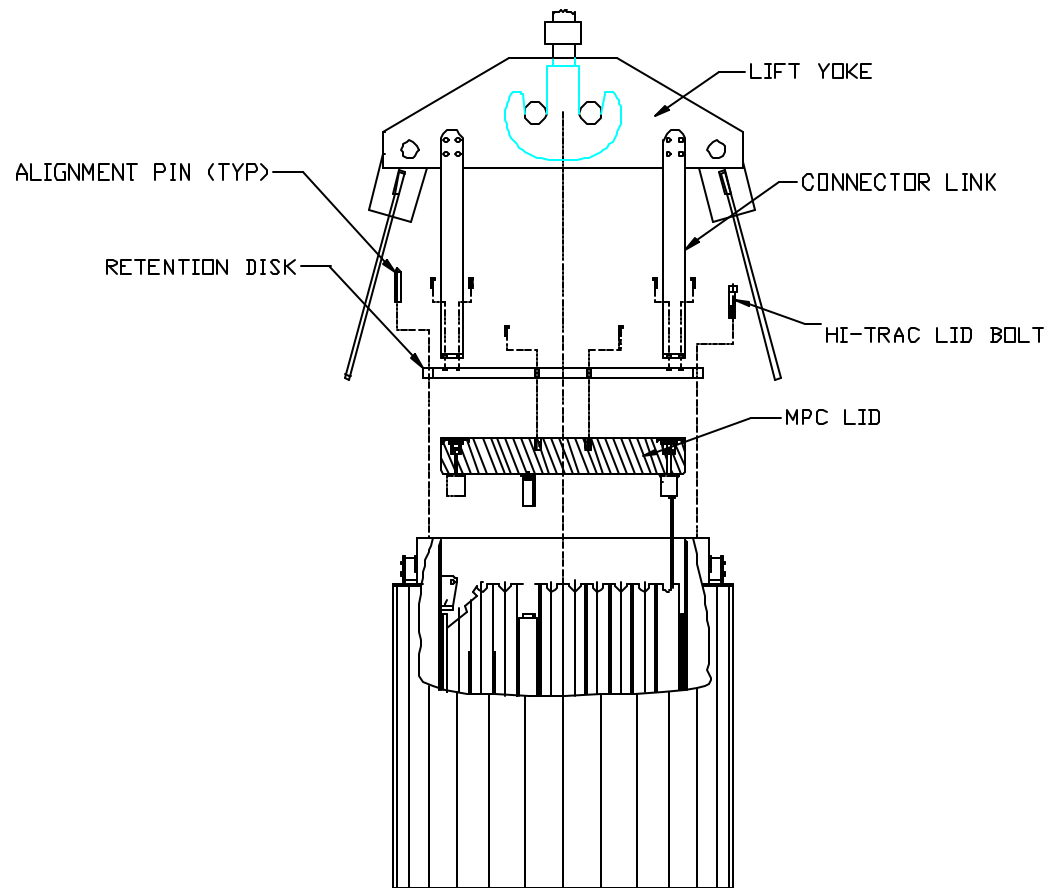
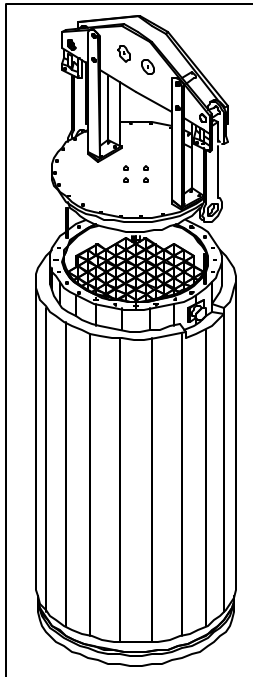
**Figure 8.1.13; Annulus Shield/Annulus Seal**

# Detail A

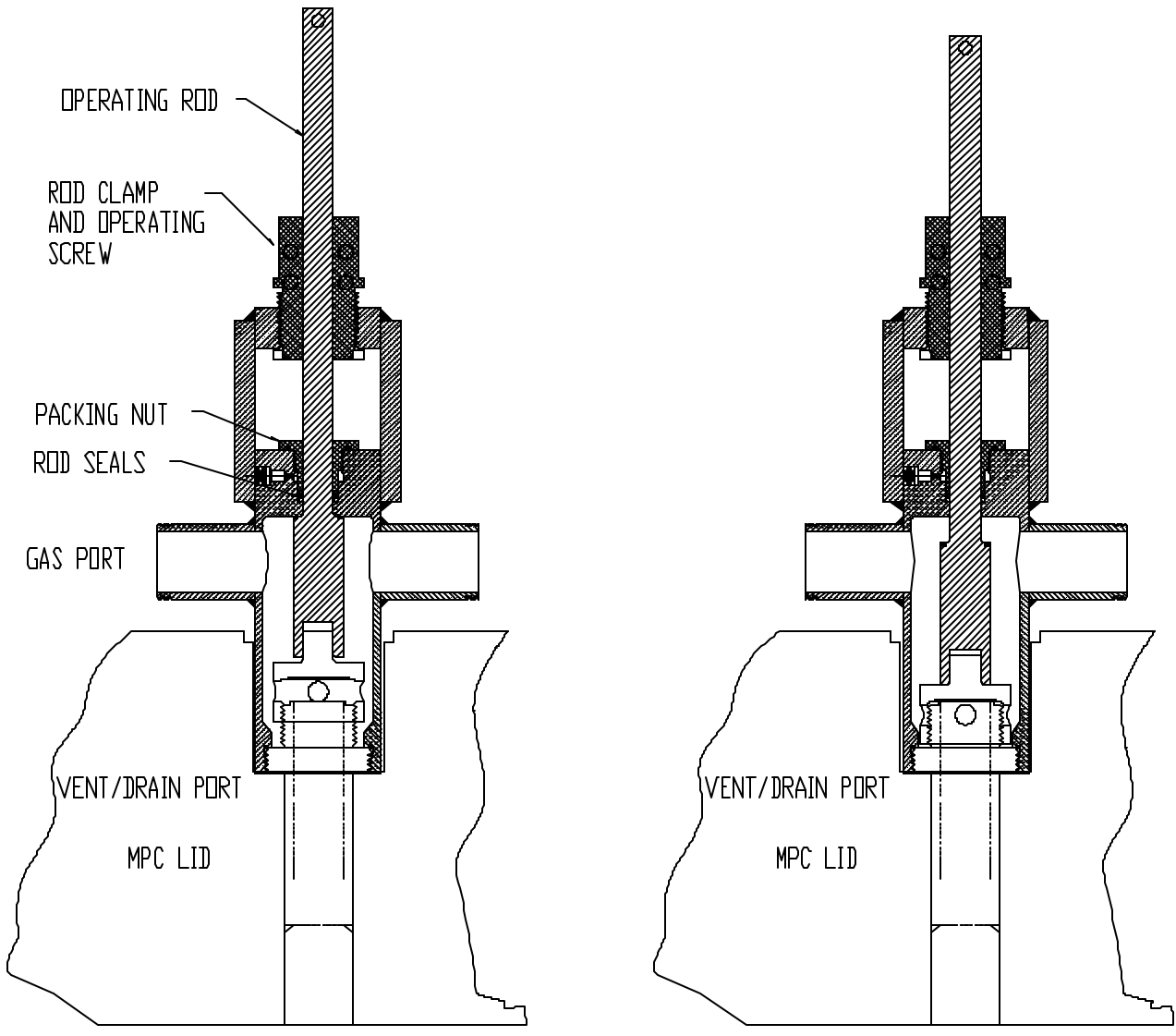


# Detail B

**Figure 8.1.14; Annulus Overpressure System**



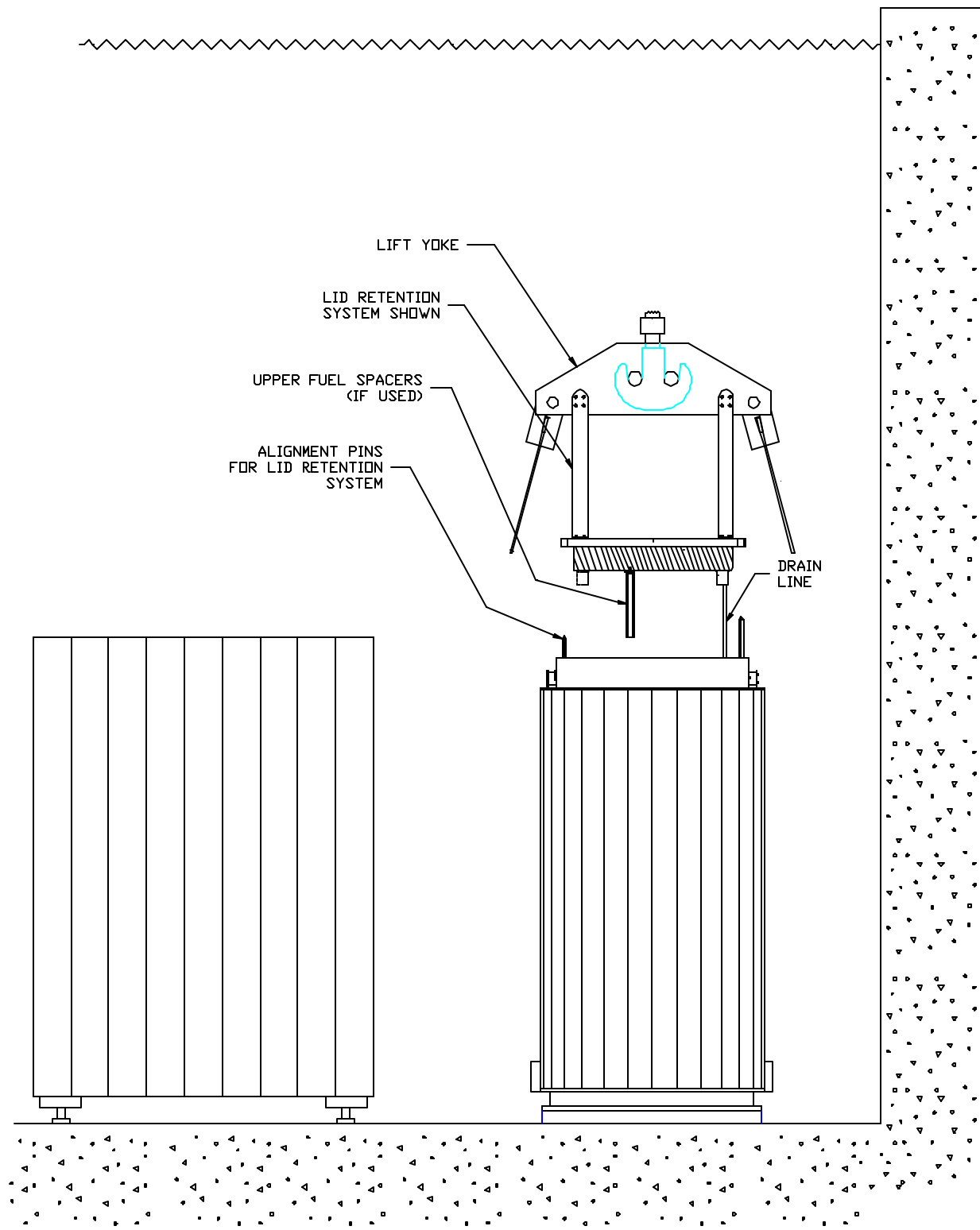
**Figure 8.1.15; HI-TRAC Lid Retention System in Exploded View**



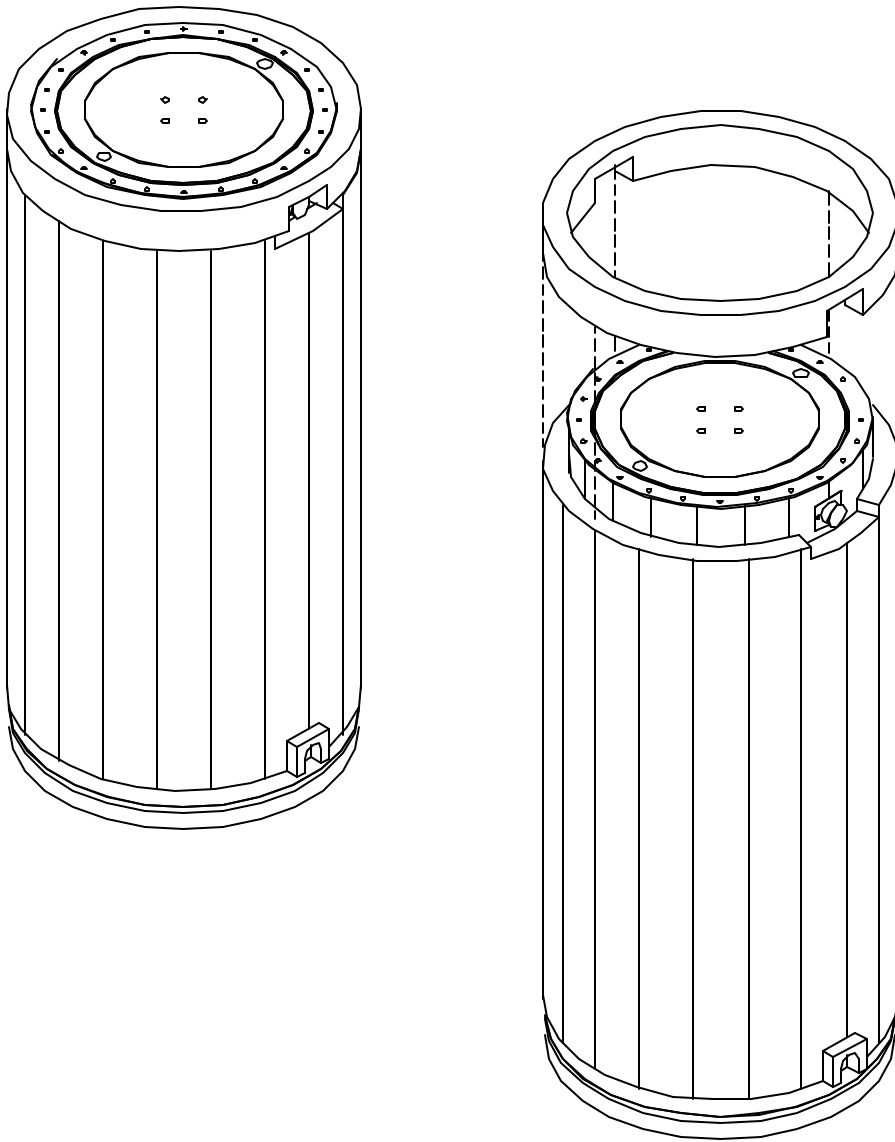
REMOVABLE VALVE  
OPERATING ASSEMBLY  
(OPEN POSITION)

REMOVABLE VALVE  
OPERATING ASSEMBLY  
(CLOSED POSITION)

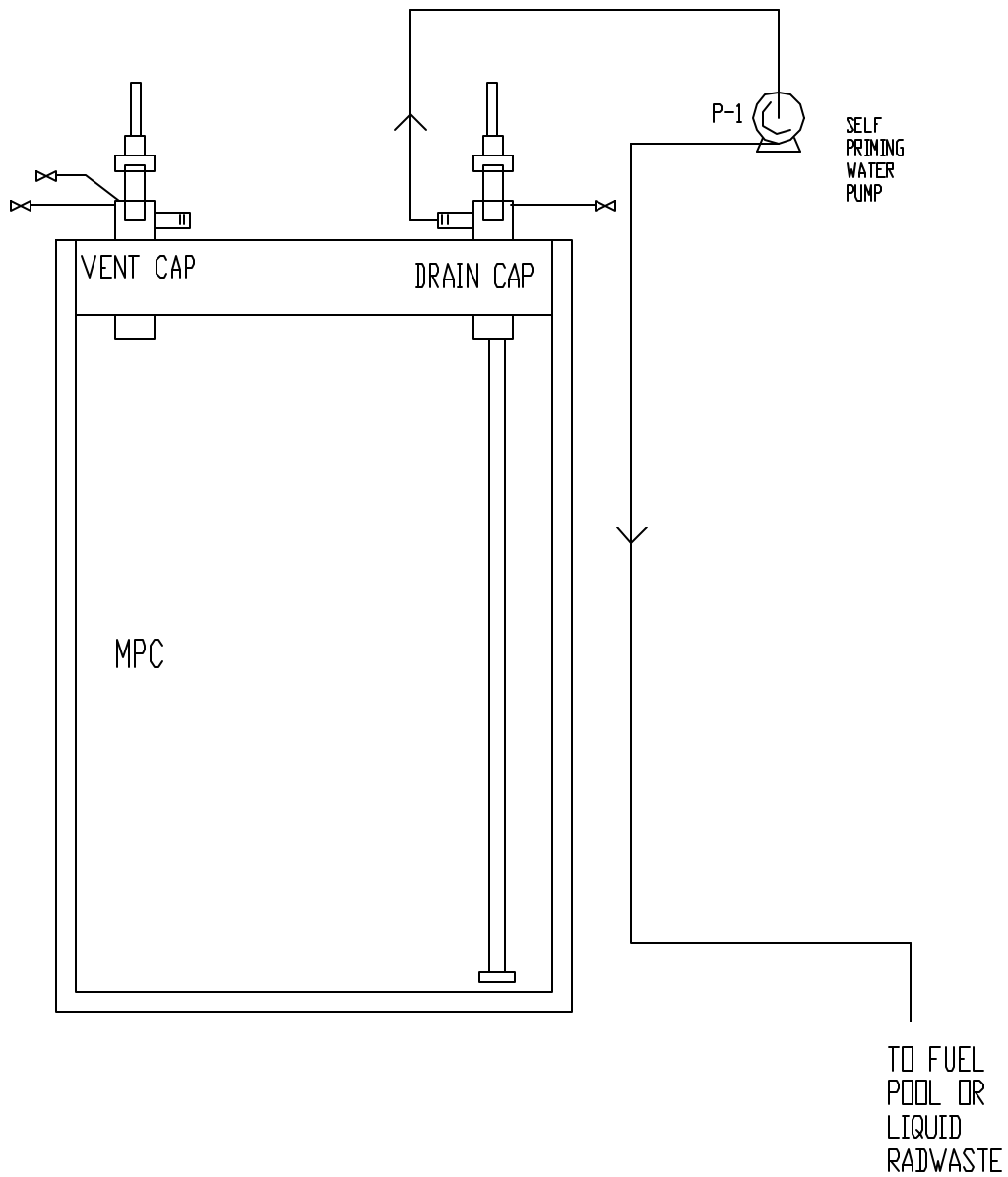
**Figure 8.1.16; MPC Vent and Drain Port RVOA Connector**



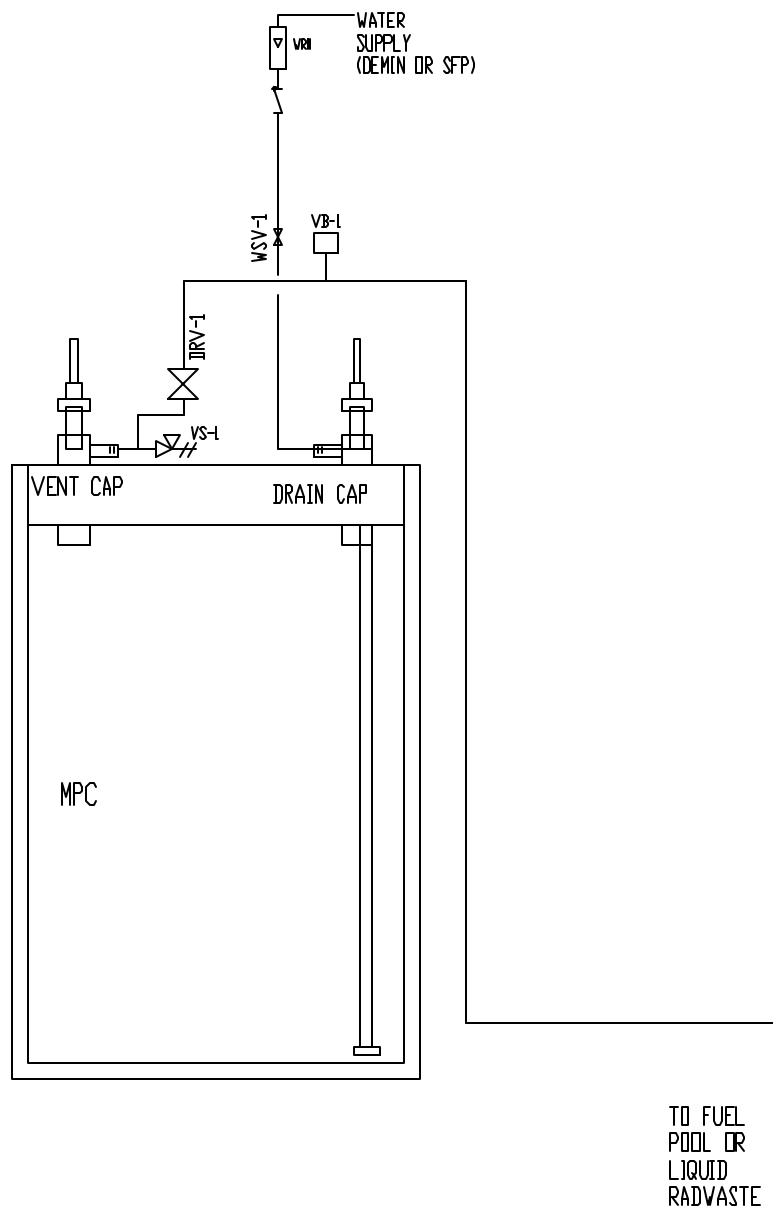
**Figure 8.1.17; Drain Line Installation**



**Figure 8.1.18; Temporary Shield Ring**

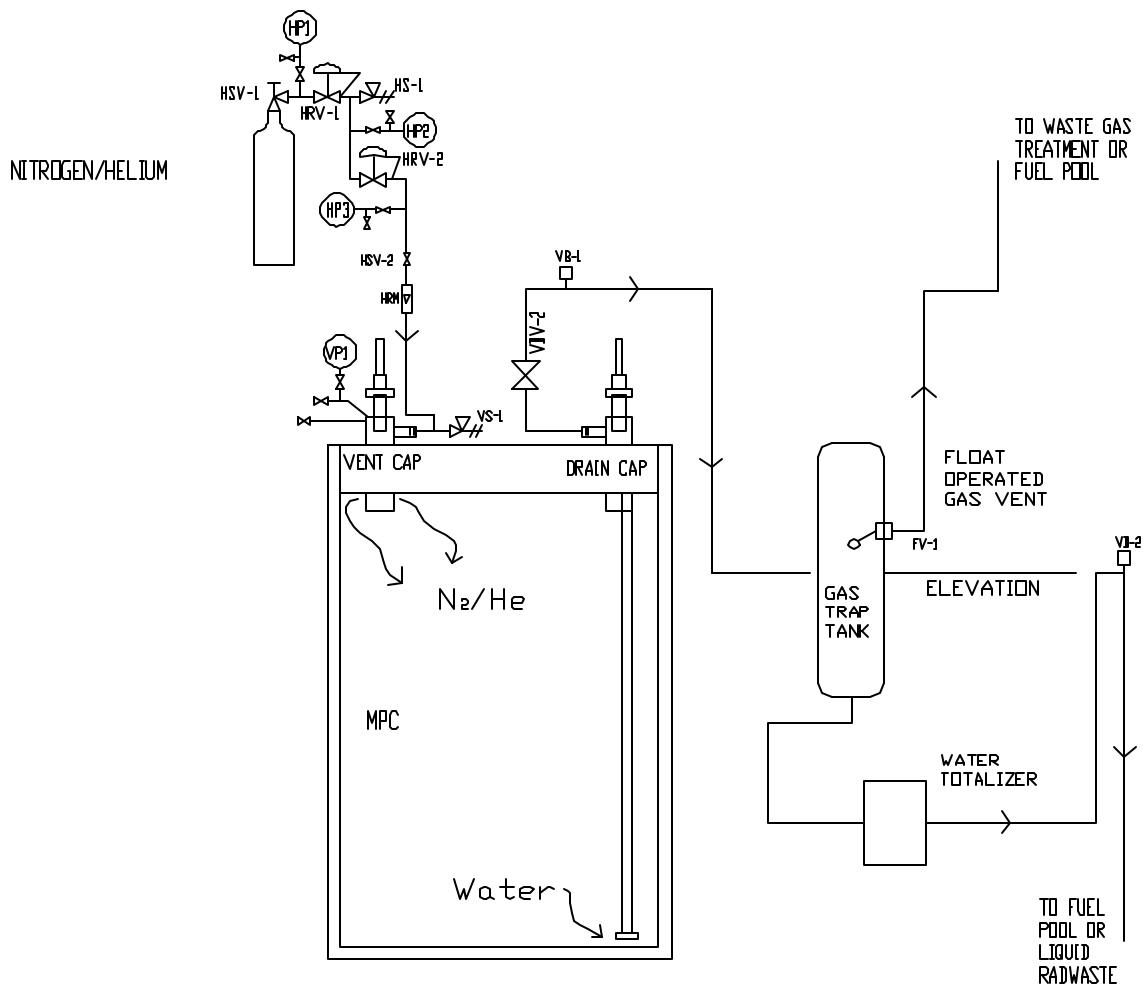


**Figure 8.1.19; MPC Water Pump-Down for MPC Lid Welding Operations, Example P&I D**

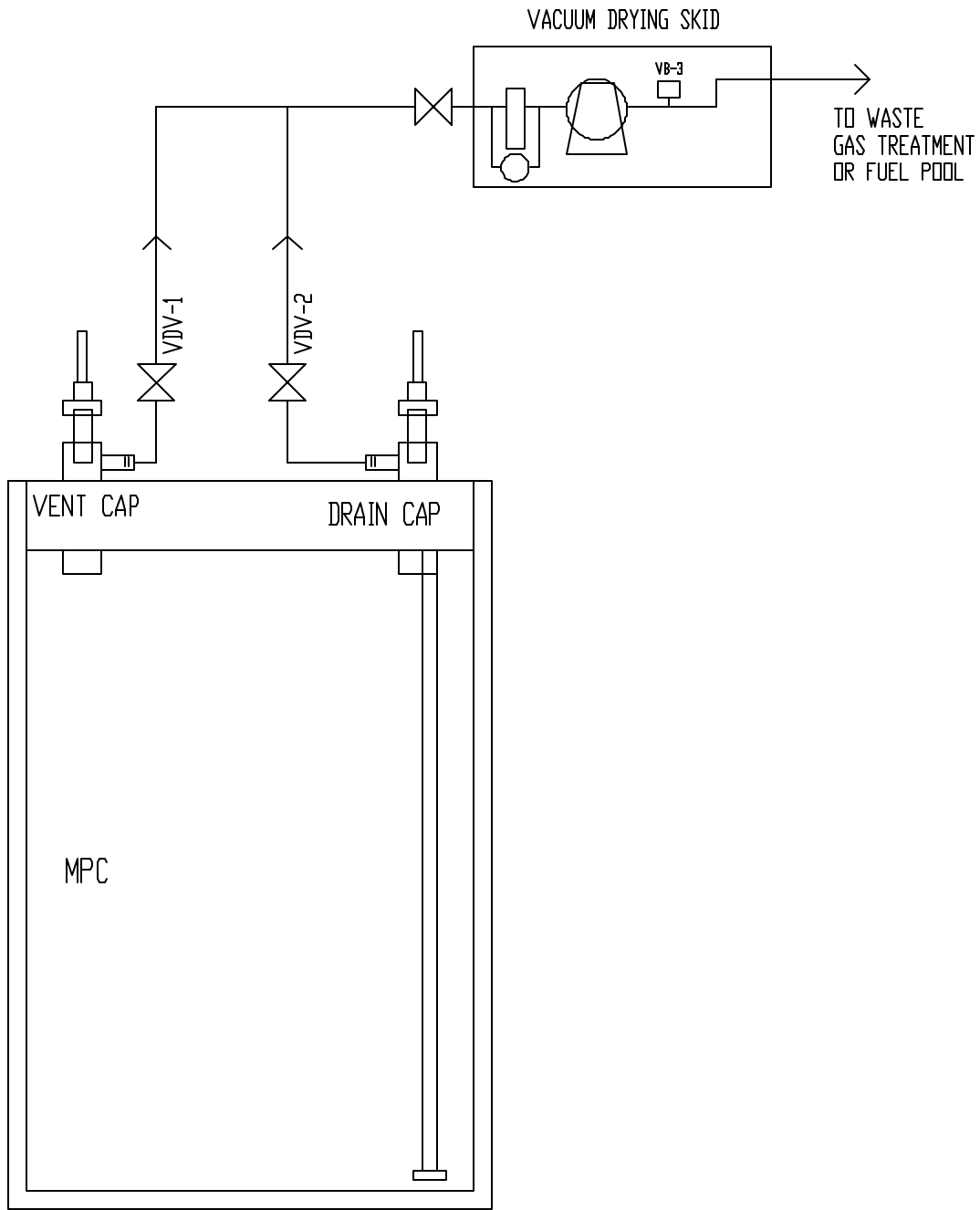


**Figure 8.1.20; MPC Air Displacement & Hydrostatic Testing, Example P&I D**

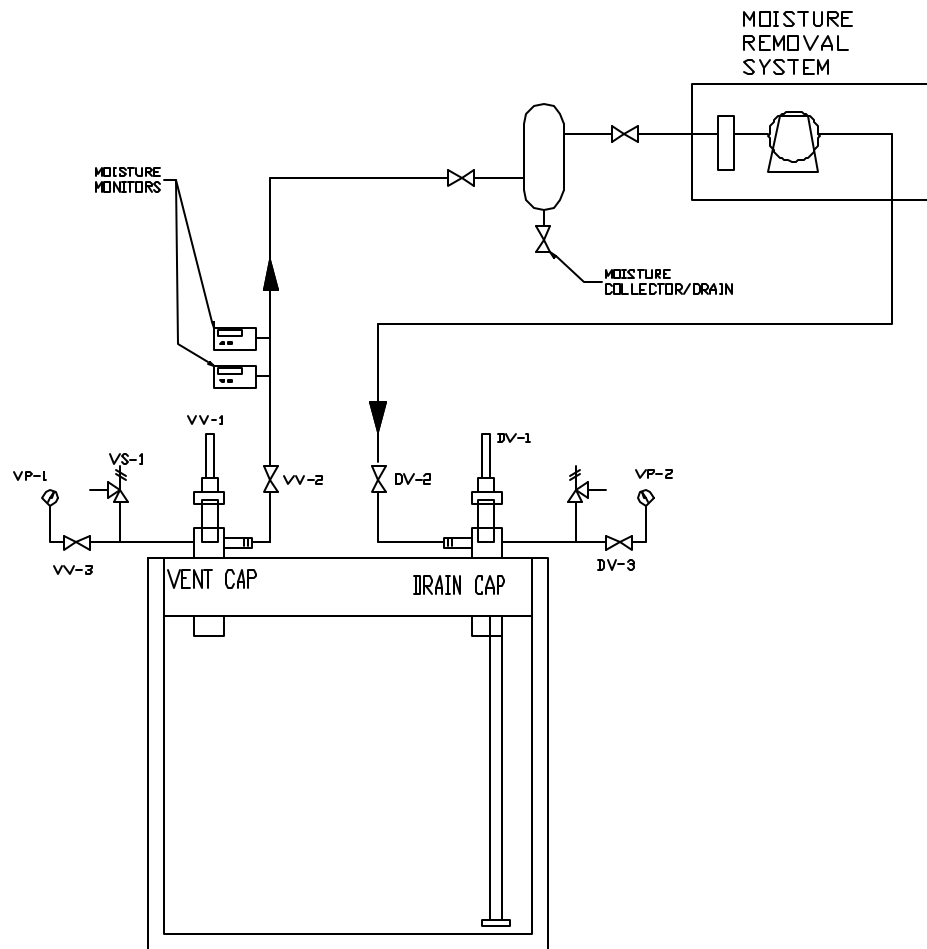




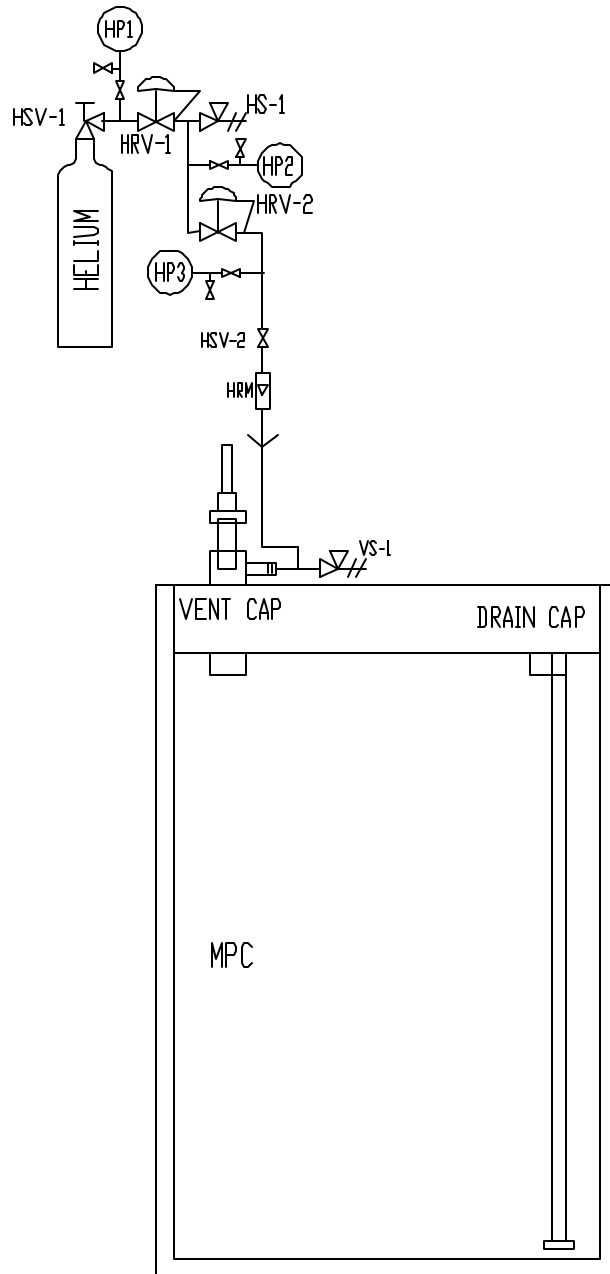
**Figure 8.1.21; MPC Blowdown, Example P&I D**



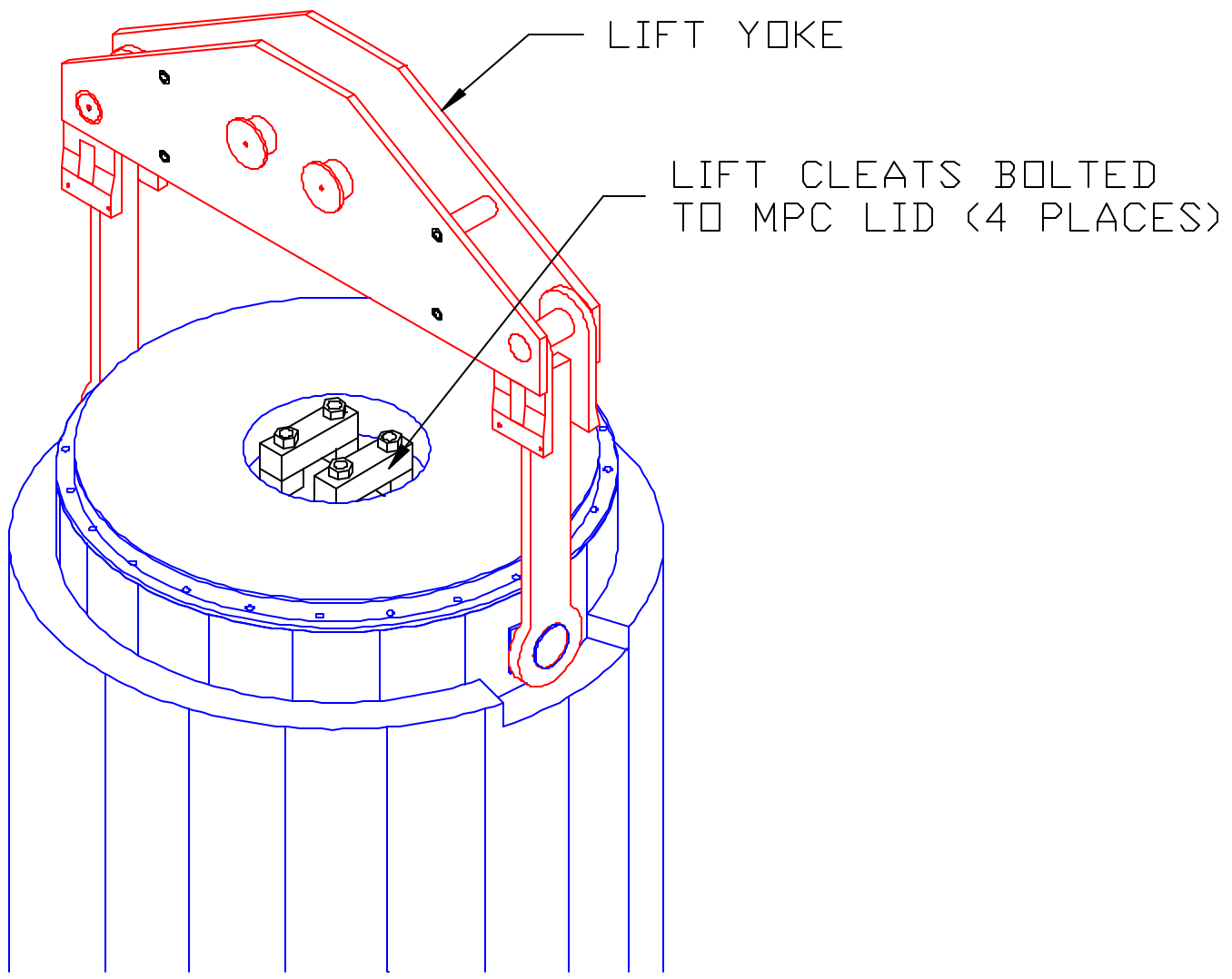
**Figure 8.1.22a; Vacuum Drying System, Example P&I D**



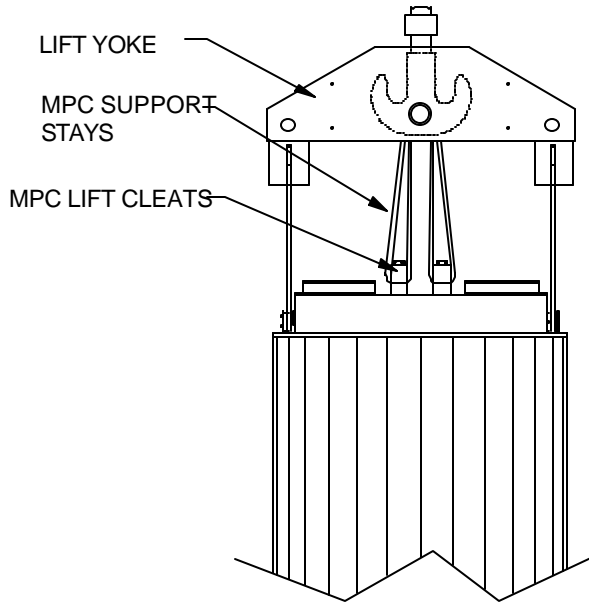
**Figure 8.1.22b; forced Helium Dehydration System, Example P&I D**



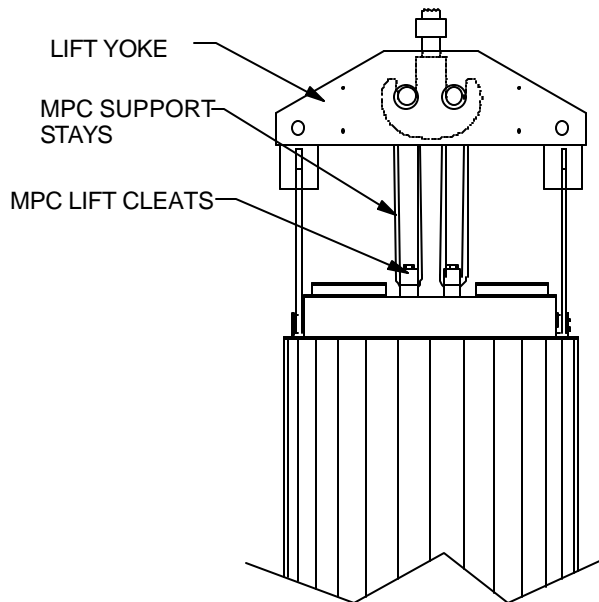
**Figure 8.1.23; Helium Backfill System, Example P&I D**



**Figure 8.1.24; MPC Lift Cleats**

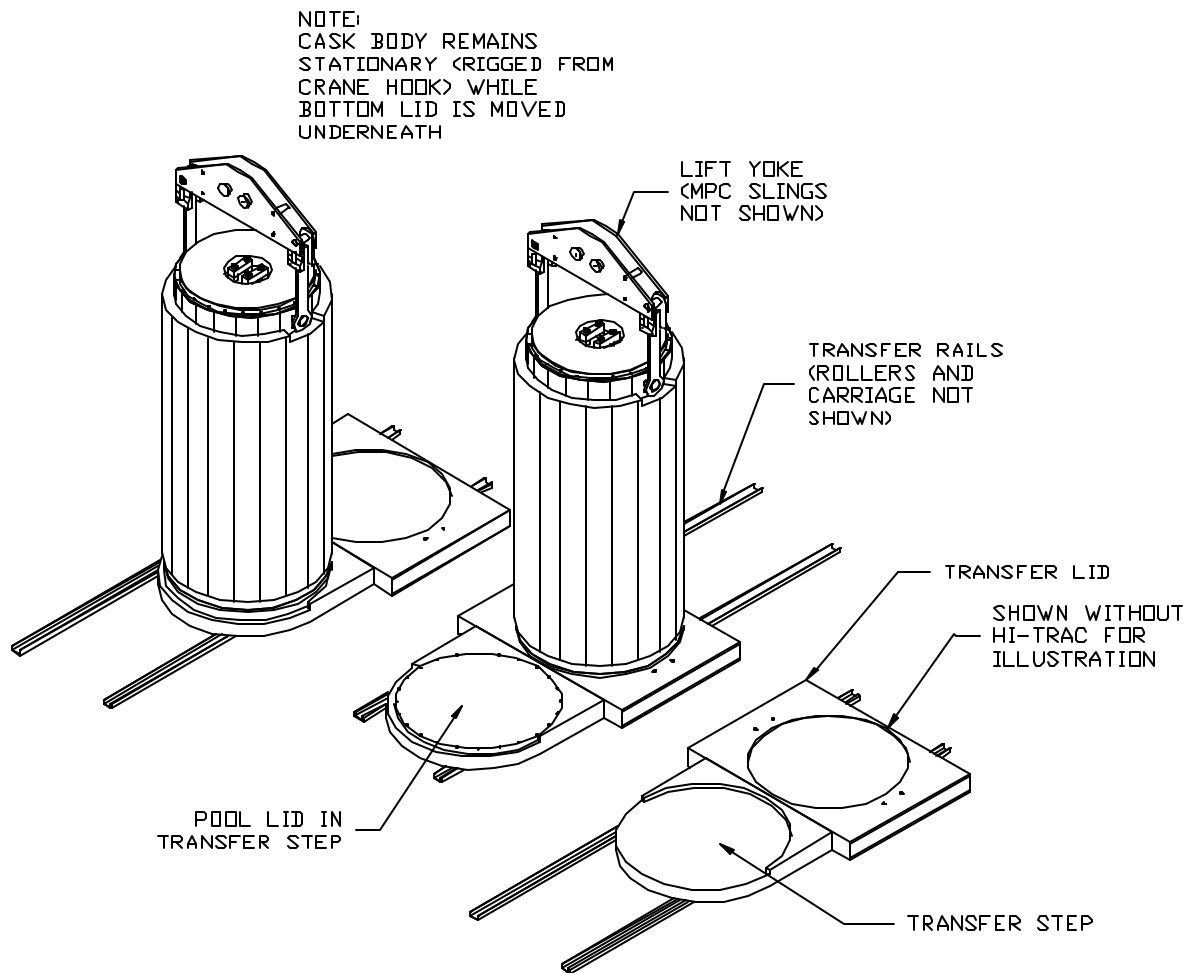


SINGLE-PIN  
ARRANGEMENT



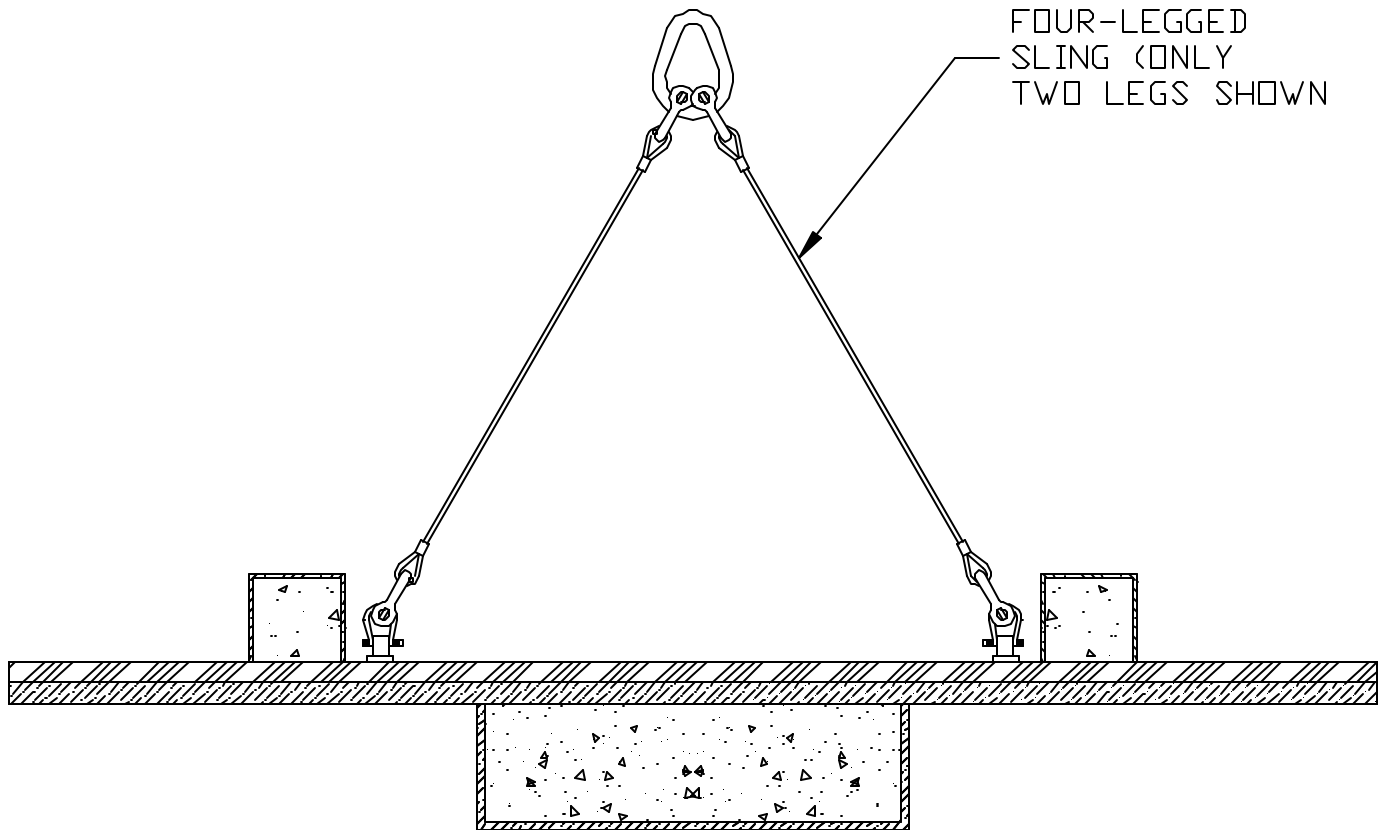
DOUBLE-PIN  
ARRANGEMENT

**Figure 8.1.25; MPC Slings**



**Figure 8.1.26; HI-TRAC Bottom Lid Replacement**

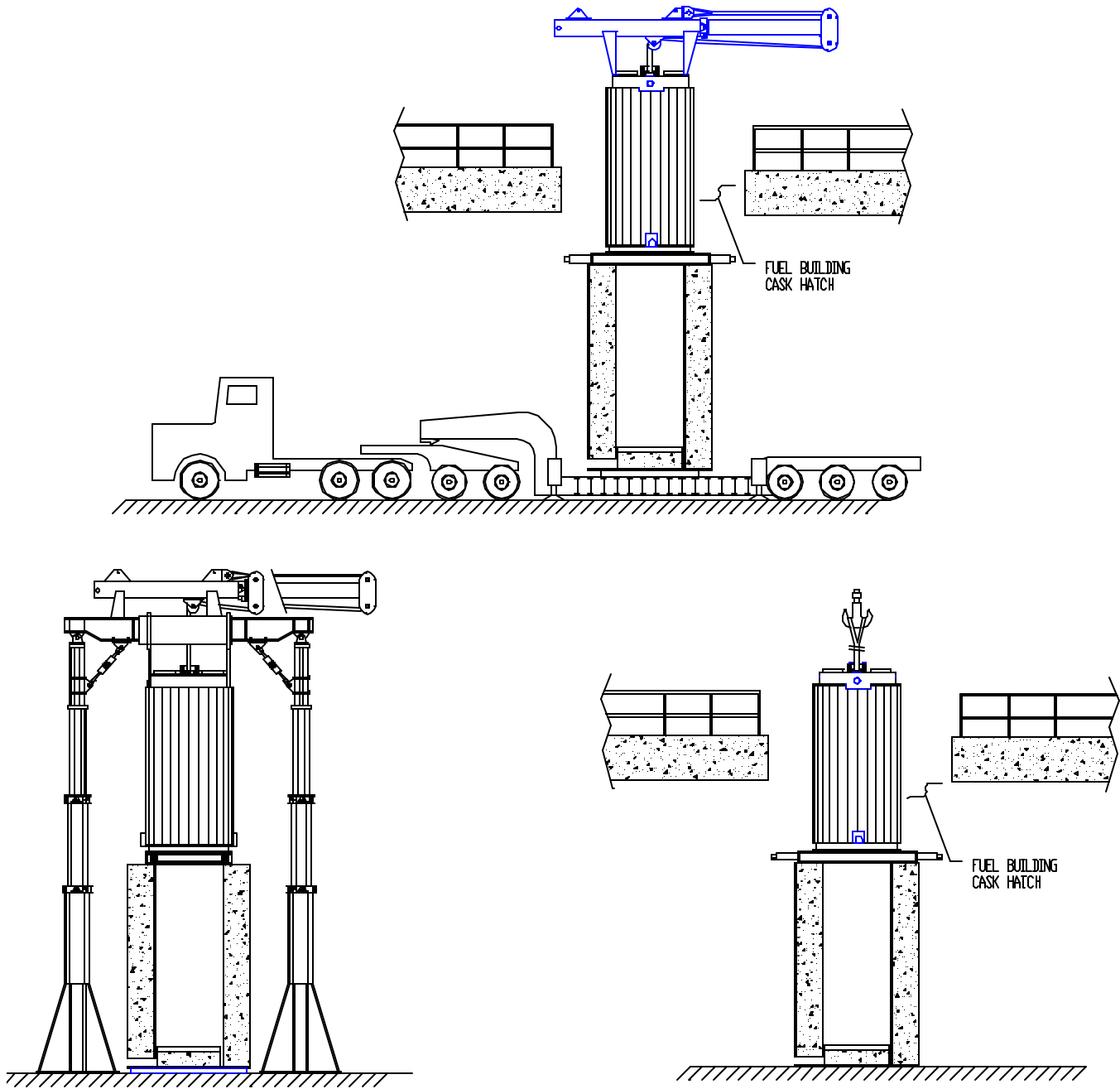
**(Not Required for HI-TRAC 125D)**



**Figure 8.1.27; HI-STORM Lid Rigging**

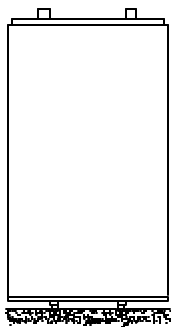
**(HI-STORM 100 Lid Shown)**



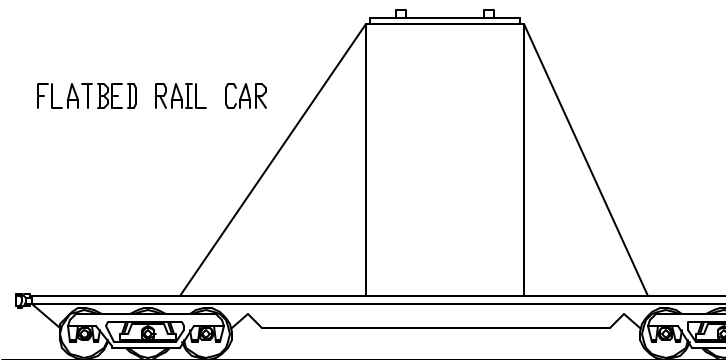


**Figure 8.1.28; Sample MPC Transfer Options  
(HI-TRAC with Transfer Lid Shown)**

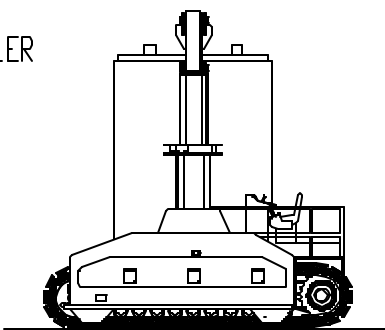
RAIL DOLLY



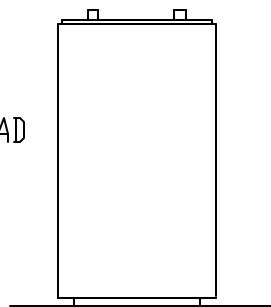
FLATBED RAIL CAR



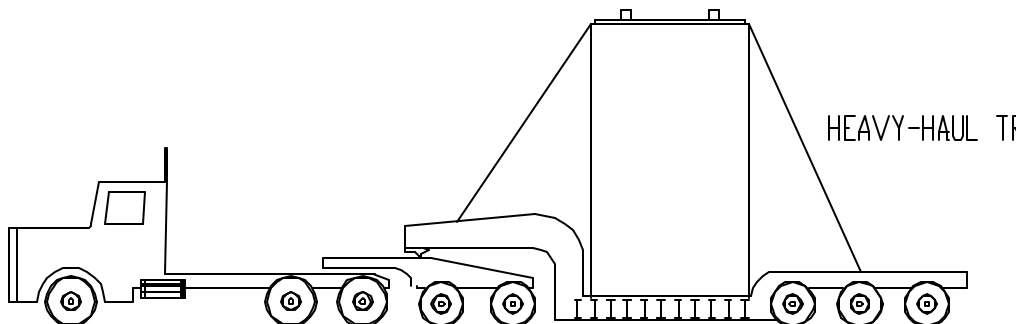
VERTICAL  
CASK CRAWLER



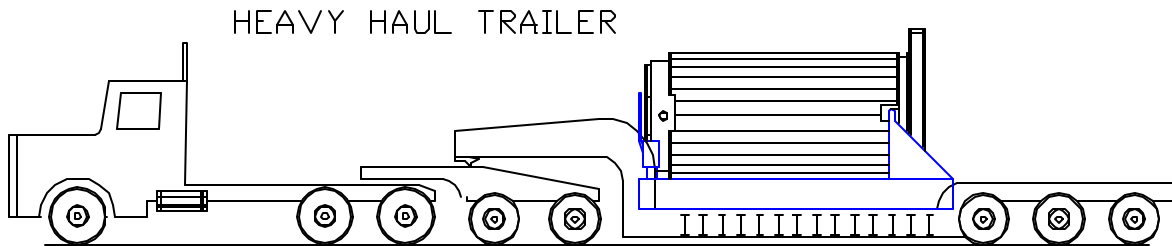
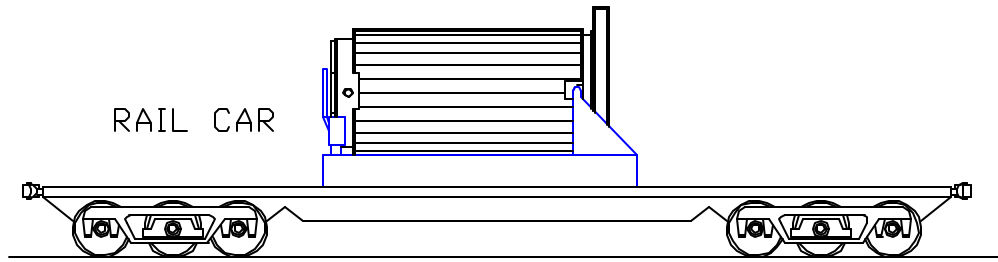
AIR PAD



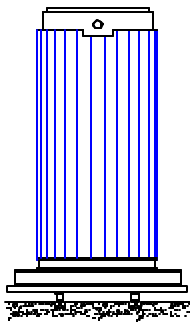
HEAVY-HAUL TRAILER



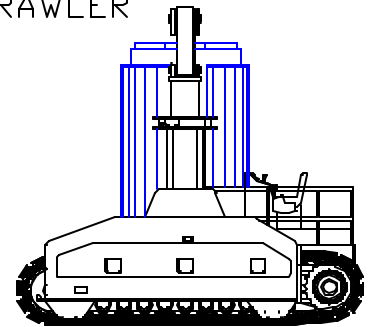
**Figure 8.1.29a; Sample HI-STORM Transfer Options**



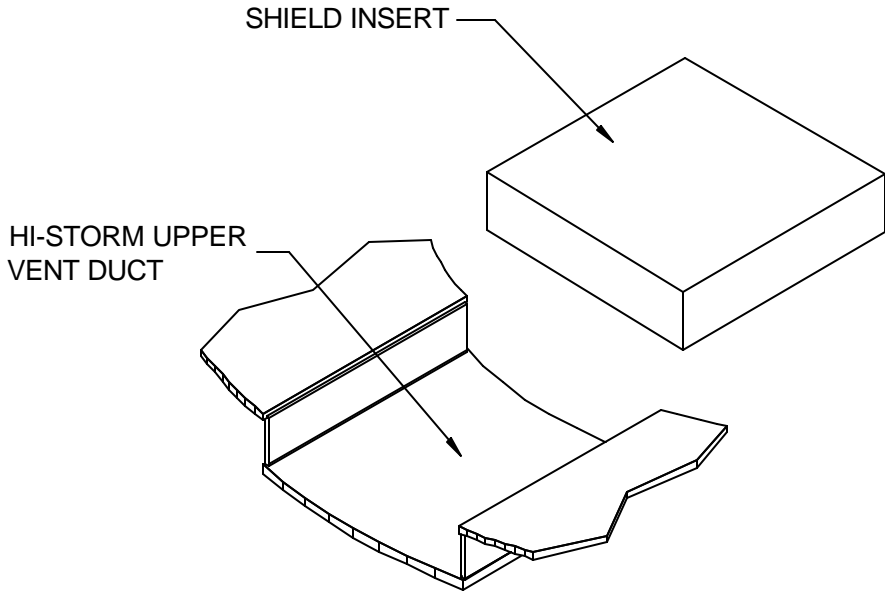
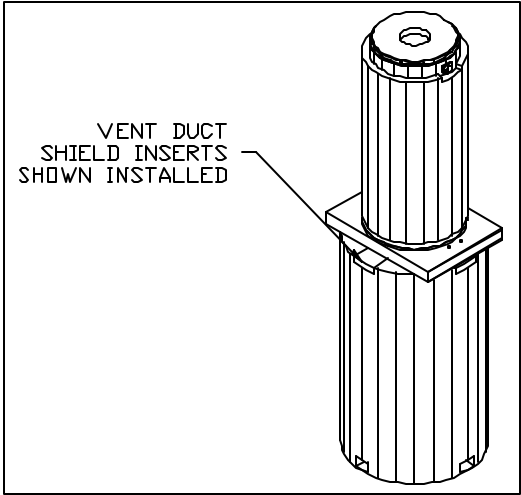
RAIL DOLLY



VERTICAL CASK CRAWLER

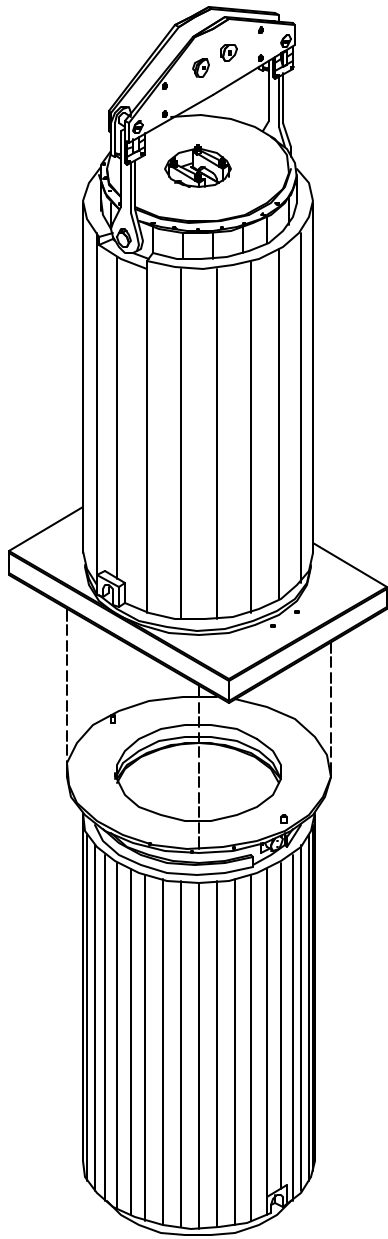


**Figure 8.1.29b; Sample HI-TRAC Transfer Options  
(HI-TRAC 100/125 Shown)**

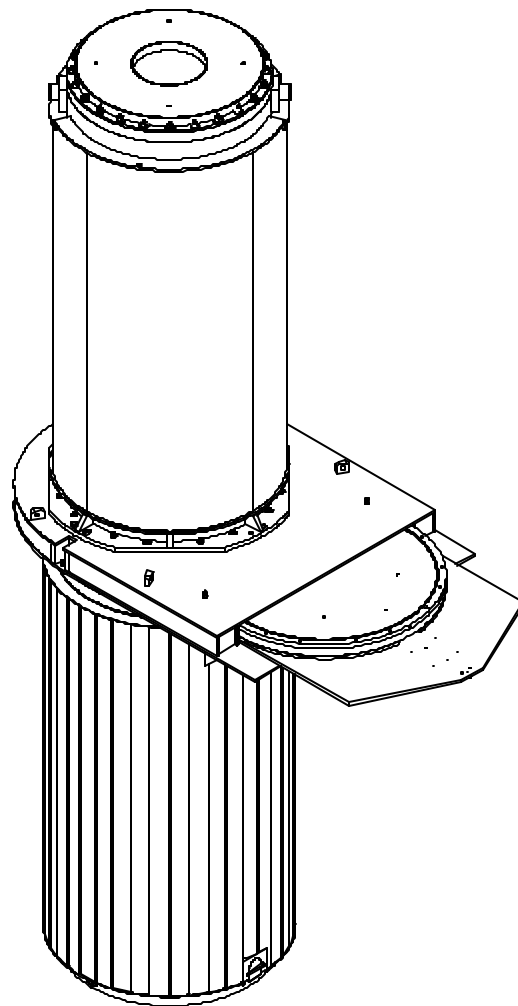


**Figure 8.1.30; Sample HI-STORM Vent Duct Shield Inserts**

**(Not Required for HI-STORM 100S)**

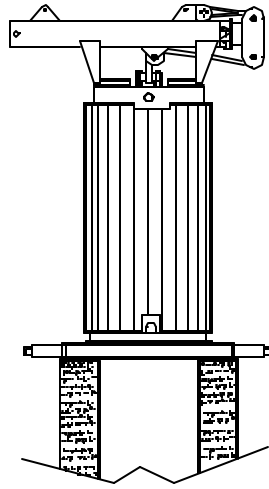


HI-TRAC 100 / 125

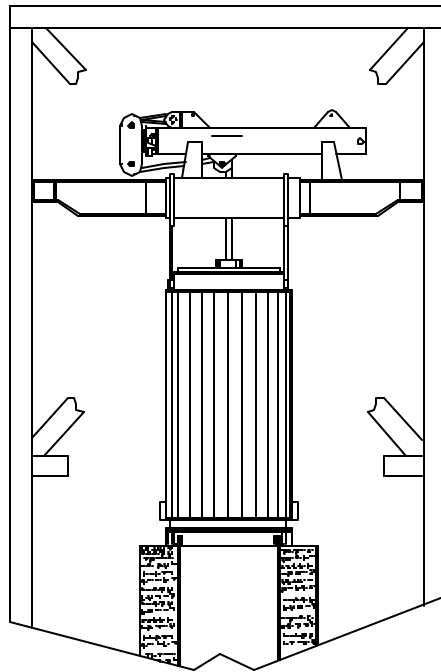


HI-TRAC 125D with Mating Device

**Figure 8.1.31; HI-TRAC Alignment Over HI-STORM**

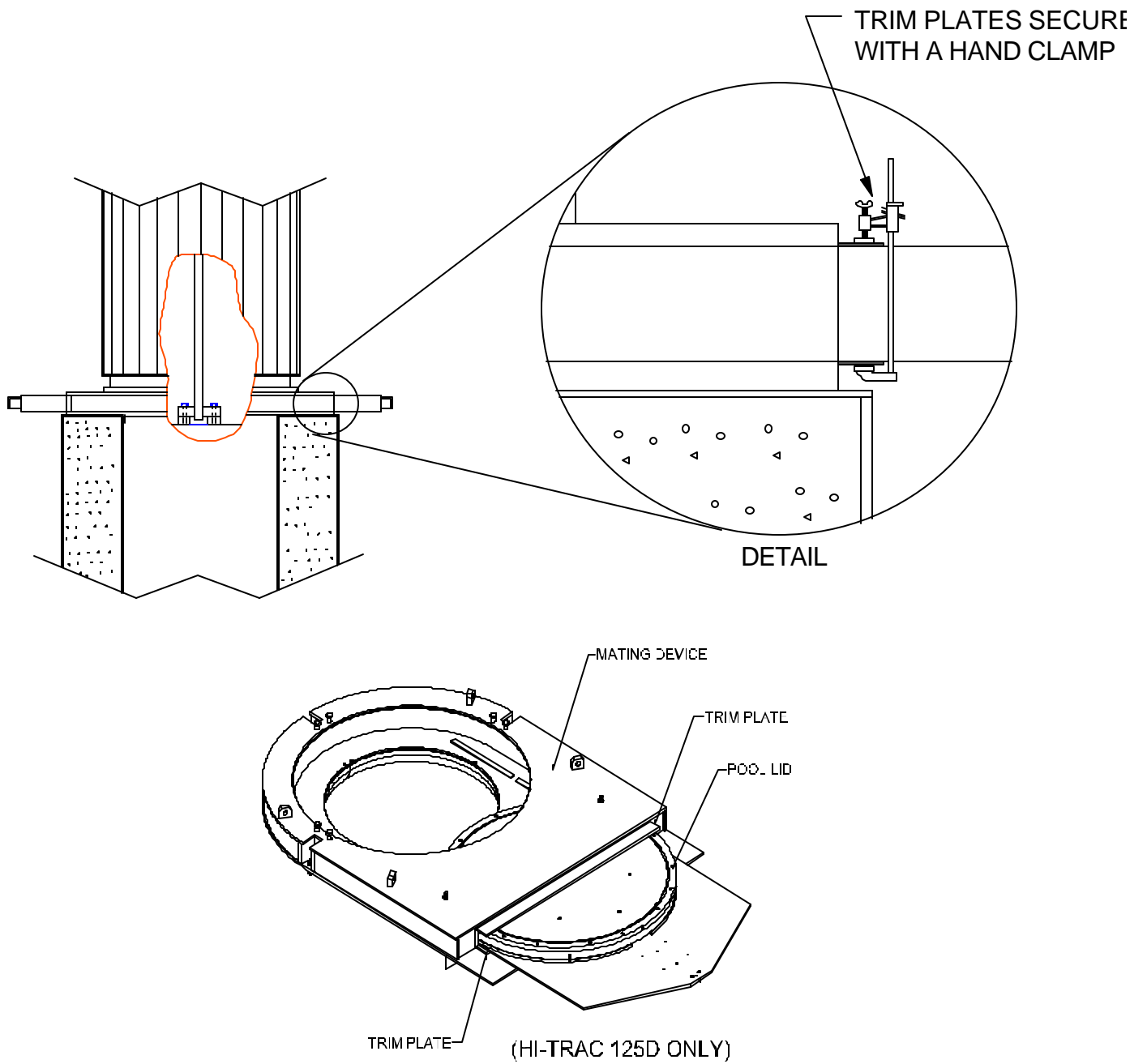


MOUNTED TO THE HI-TRAC LID

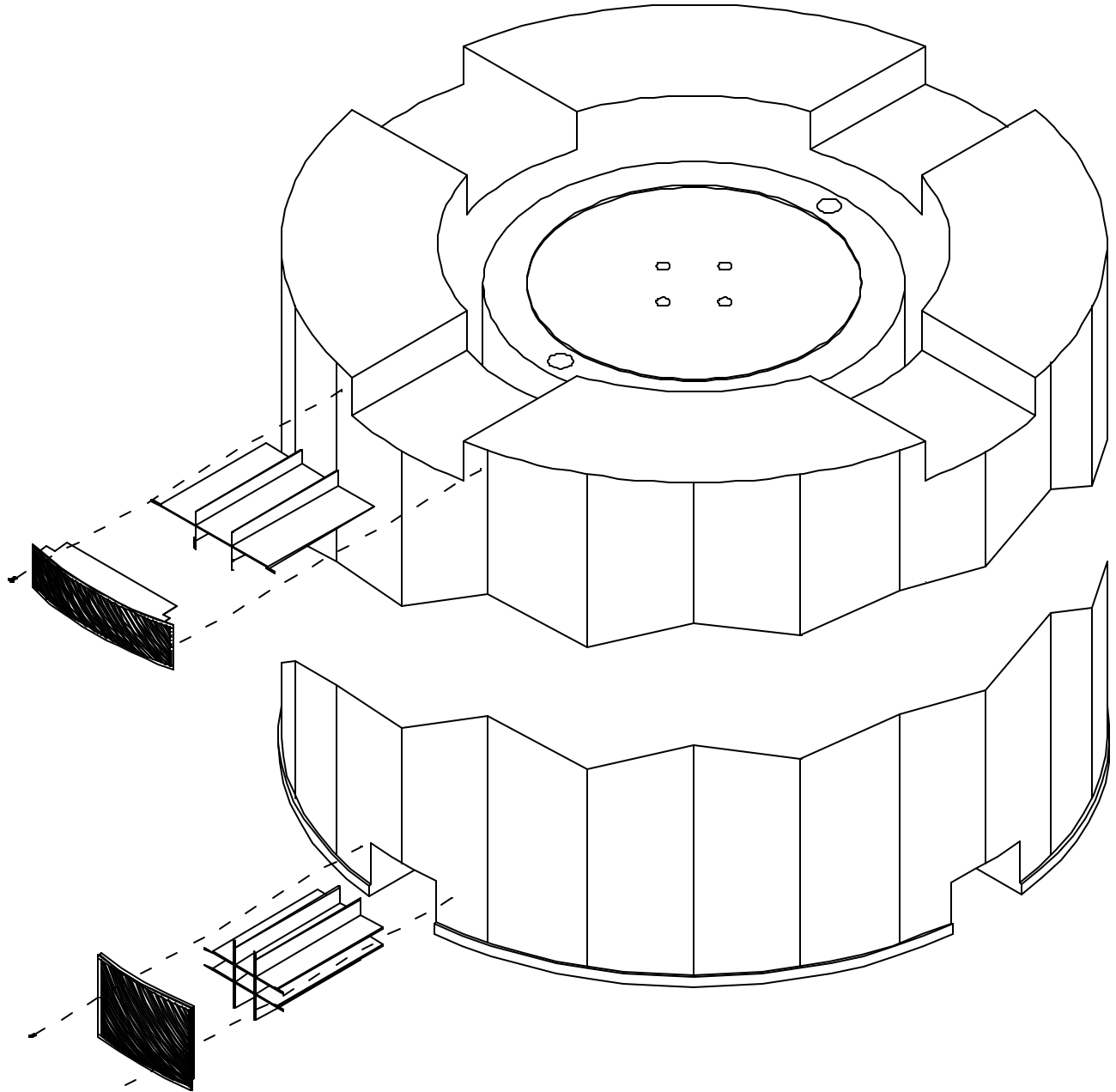


MOUNTED TO A CASK TRANSFER GANTRY

**Figure 8.1.32; Examples of an MPC Downloader**



**Figure 8.1.33; Trim Plate Locations**

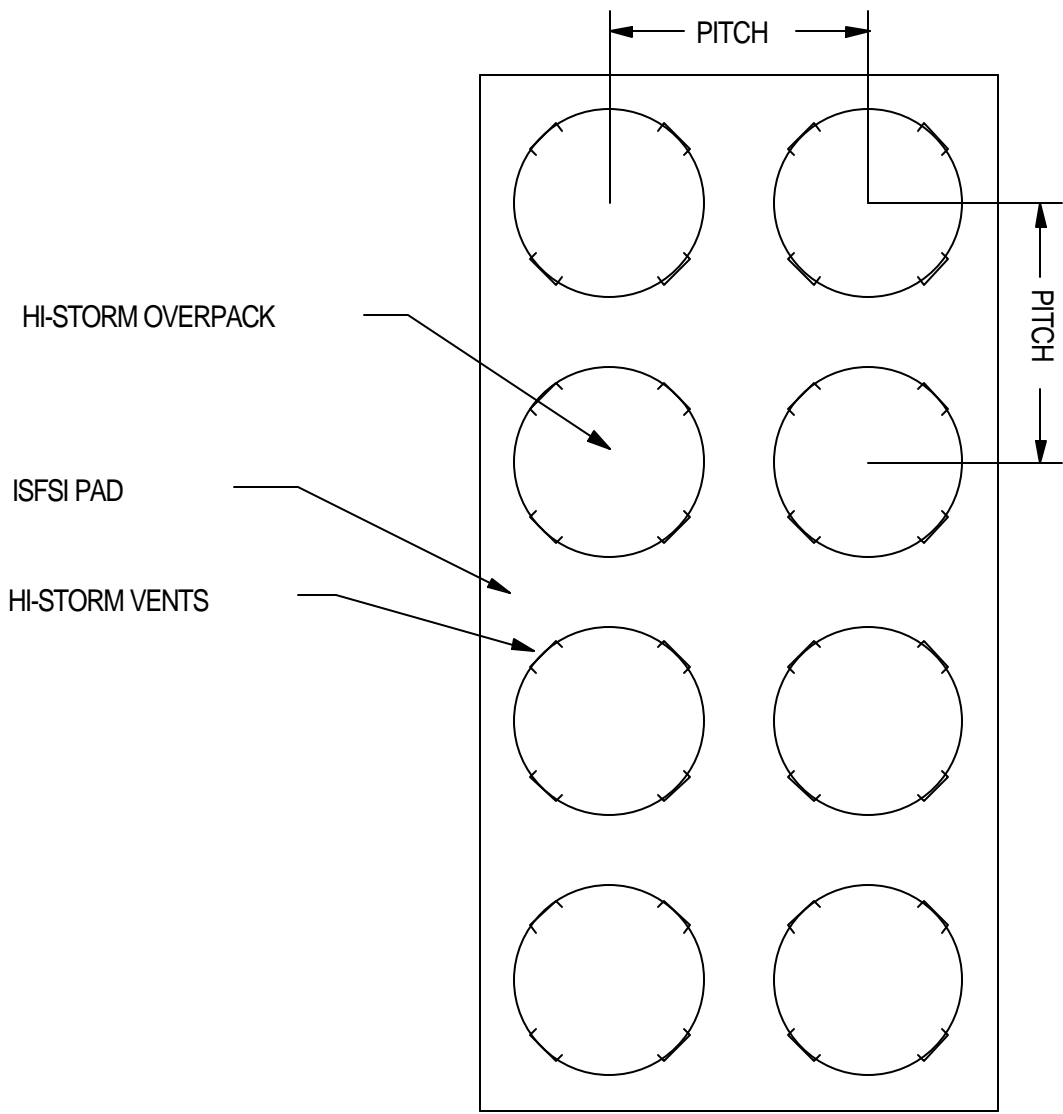


**Figure 8.1.34a; Typical HI-STORM Vent Screen and Gamma Shield Cross Plate Installation**

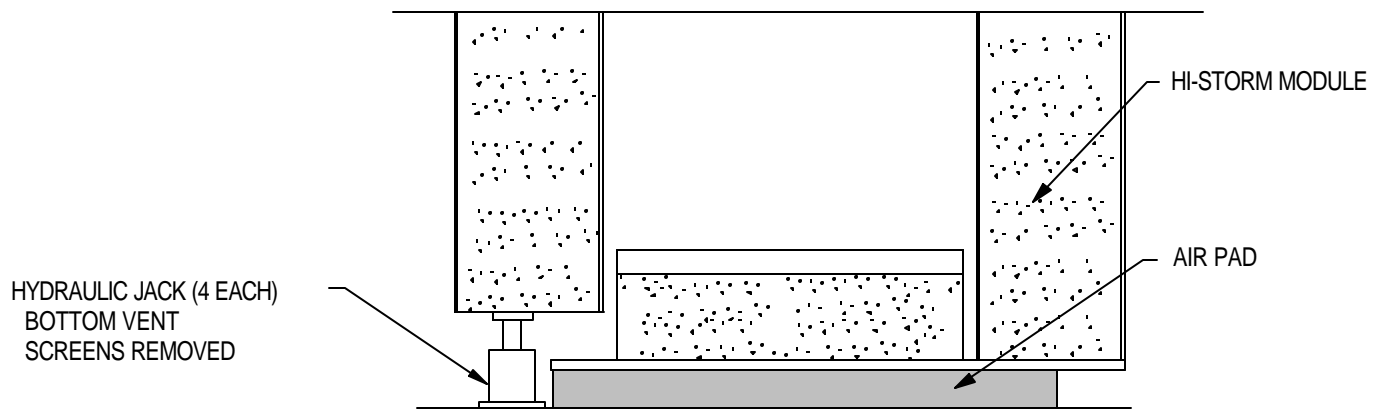


**Figure 8.1.34b**

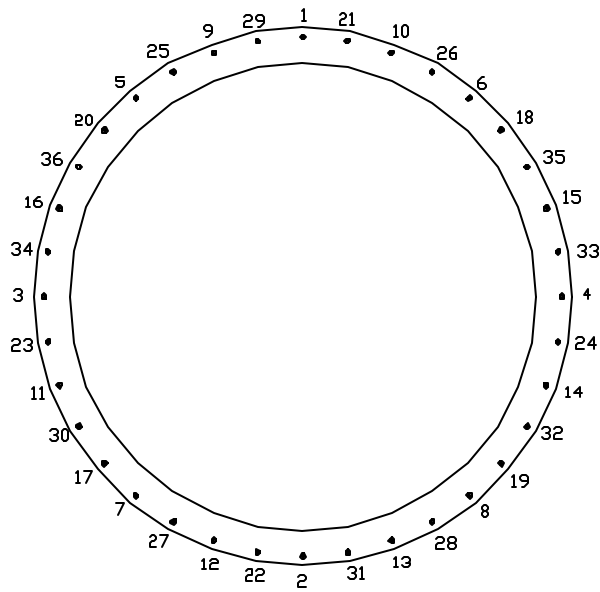
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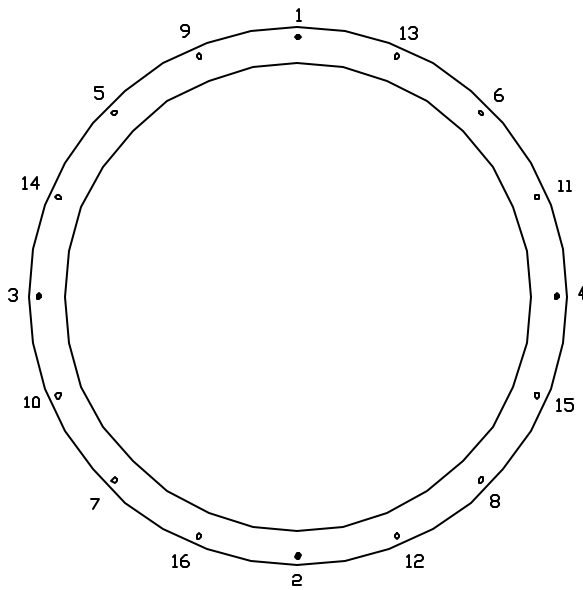
**Figure 8.1.35; HI-STORM Placement of the ISFSI Pad**



**Figure 8.1.36; HI-STORM Jacking**



HI-TRAC 100 / 125  
 BOTTOM LID  
 (POOL LID OR  
 TRANSFER LID)



HI-TRAC 125D  
 POOL LID

**Figure 8.1.37; HI-TRAC Lid Bolt Torquing Pattern**

## 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 (~~See Technical Specification LCO 3.1.2~~) 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 in accordance with Technical Specification SR 3.1.2.1.~~
- 8.2.2 ISFSI Security Operations shall be performed in accordance with the approved site security program plan.

## 8.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

### 8.3.1 Overview of HI-STORM 100 System Unloading Operations

**ALARA Note:**

The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. Figure 8.3.1 shows a flow diagram of the HI-STORM unloading operations. Figure 8.3.2 illustrates the major HI-STORM unloading operations.

Refer to the boxes of Figure 8.3.2 for the following description. The MPC is recovered from HI-STORM either at the ISFSI or the fuel building using the same methodologies as described in Section 8.1 (Box 1). The HI-STORM lid is removed, the vent duct shield inserts are installed, the alignment device (or mating device with pool lid for HI-TRAC 125D) is positioned, and the MPC lift cleats are attached to the MPC. The exit vent screens and gamma shield cross plates are removed as necessary. MPC slings are attached to the MPC lift cleat and positioned on the MPC lid. HI-TRAC is positioned on top of HI-STORM (Box 2) and the slings are brought through the HI-TRAC top lid. The MPC is raised into HI-TRAC, the HI-TRAC doors (or mating device drawer) are closed and the locking pins are installed. If the mating device and HI-TRAC 125D are used, the pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of HI-STORM. If the HI-TRAC 125D is not used, the HI-TRAC is positioned in the transfer slide and the transfer lid is replaced with the pool lid (Box 3) using the same methodology as with the loading operations.

*If the MPC contains high burn-up fuel, a Supplemental Cooling System (SCS) (if required) is connected to the HI-TRAC annulus following transfer from the HI-STORM to the HI-TRAC and used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits. HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC slings, MPC lift cleats and top lid are removed<sup>1</sup> (Box 4). The temporary shield ring is installed on the HI-TRAC upper section and filled with plant demineralized water. The HI-TRAC top lid is removed and a water flush is performed on the annulus. Water is fed into the annulus through the drain port and allowed to cool the MPC shell. After a predetermined period (based on the fuel conditions), cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid. The weld removal system is installed (Box 7) and the MPC vent and drain ports are accessed (Box 5). The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample*

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<sup>1</sup> Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is cooled using the cool-down system to reduce the MPC internal temperature to allow water flooding (Box 6). The cool-down process gradually reduces the cladding temperature to a point where the MPC may be flooded with water without thermally shocking the fuel assemblies or over-pressurizing the MPC from the formation of steam (~~See Technical Specification LCO 3.1.3~~). Following the fuel cool-down, the MPC is filled with water (*borated as required*) and the supplemental cooling is terminated (if used). The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place (Box 7).

The top surfaces of the HI-TRAC and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or lid retention system and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed (Boxes 8 and 9). All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris and crud (Box 10). HI-TRAC and MPC are returned to the designated preparation area (Box 11) where the MPC water is pumped back into the spent fuel pool or liquid radwaste facility. The annulus water is drained and the MPC and overpack are decontaminated (Box 12 and 13).

### 8.3.2 HI-STORM Recovery from Storage

**Note:**

The MPC transfer may be performed using the MPC downloader or the overhead crane.

1. Recover the MPC from HI-STORM as follows:
  - a. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met. ~~See Technical Specifications for the on-site lifting requirements.~~
  - b. Transfer HI-STORM to the fuel building or site designated location for the MPC transfer.
  - c. Position HI-STORM under the lifting device.
  - d. Remove the HI-STORM lid nuts, washers and studs.
  - e. Remove the HI-STORM lid lifting hole plugs and install the lid lifting sling. See Figure 8.1.27.

**Note:**

The specific sequence for vent screen, temperature element, and gamma shield cross plate removal may vary based on the mode(s) or transport.

- f. Remove the HI-STORM exit vent screens, temperature elements and gamma shield cross plates. See Figure 8.1.34a and b.

**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) for the HI-STORM lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

- g. Remove the HI-STORM lid. See Figure 8.1.27.
  - h. Install the alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (HI-STORM 100 only). See Figure 8.1.30.
  - i. Deleted.
  - j. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings to the MPC lid. See Table 8.1.5 for torque requirements.
  - k. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for rigging. See Table 8.1.5 for torque requirements.
  - l. Deleted.
2. If necessary, configure HI-TRAC with the transfer lid (Not required for HI-TRAC 125D):

**ALARA Warning:**

The bottom lid replacement as described below may only be performed on an empty (i.e., no MPC) HI-TRAC.

- a. Position HI-TRAC vertically adjacent to the transfer lid. See Section 8.1.2.
  - b. Remove the bottom lid bolts and plates and store them temporarily.
  - c. Raise the empty HI-TRAC and position it on top of the transfer lid.
  - d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - e. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.
3. At the site's discretion, perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

**Note:**

If the HI-TRAC is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

4. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.
  5. Engage the lift yoke to the HI-TRAC lifting trunnions.



6. Align HI-TRAC over HI-STORM and mate the overpacks. See Figure 8.1.31.
7. If necessary, install the MPC downloader.
8. Remove the transfer lid (or mating device) locking pins and open the doors (mating device drawer).
9. At the user's discretion, install trim plates to cover the gap above and below the door (drawer for 125D). The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
10. Attach the ends of the MPC sling to the lifting device or MPC downloader. See Figure 8.1.32.

**ALARA Warning:**

If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

**Caution:**

*Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance. For MPCs containing high burn-up fuel, the Supplemental Cooling System (SCS) (if required) is used to prevent fuel cladding temperatures from exceeding ISG-11 Rev. 3 limits. Operation of the SCS typically begins as soon as the MPC is placed in the HI-TRAC and continues until MPC cool-down and re-flooding operations have commenced. Staging and check-out of the SCS shall be completed prior to transferring the MPC to the HI-TRAC to minimize the time required to begin its operation.*

11. Raise the MPC into HI-TRAC.
12. Verify the MPC is in the full-up position.
13. Close the HI-TRAC doors (or mating device drawer) and install the door locking pins.
14. For the HI-TRAC 125D, bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.
15. Lower the MPC onto the transfer lid doors (or pool lid for 125D).
16. Disconnect the slings from the MPC lift cleats.

**Note:**

*For the HI-TRAC 100 and HI-TRAC 125, operation of the SCS may need to be postponed until the pool lid is in place on the HI-TRAC. In any event, supplemental cooling shall begin before time limits established by the canister thermal evaluation are exceeded.*

**Warning:**

*At the start of SCS operations, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of filling the annulus. Water addition should be performed in a slow and controlled manner until water steam generation has ceased.*

17. *If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1). Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.*
18. *If necessary, remove the MPC downloader from the top of HI-TRAC.*
- ~~18~~.19. Remove HI-TRAC from the top of HI-STORM.

### 8.3.3 Preparation for Unloading:

1. Replace the ~~transfer pool~~ lid with the transfer pool lid as follows (Not required for HI-TRAC 125D):
  - a. Lower the lift yoke and attach the MPC slings between the lift cleats and the lift yoke. See Figure 8.1.25.
  - b. Engage the lift yoke to the HI-TRAC lifting trunnions.
  - c. Deleted.
  - d. Raise HI-TRAC and position the transfer lid approximately one inch above the transfer step. See Figure 8.1.26.
  - e. Raise the transfer slide carriage so the transfer carriage is supporting the transfer lid bottom. Remove the transfer lid bolts and store them temporarily.

**ALARA Warning:**

Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

- f. Lower the transfer carriage and position the pool lid under HI-TRAC.
  - g. Raise the transfer slide carriage to place the pool lid against the HI-TRAC bottom lid bolting flange.
  - h. Inspect the bottom lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - i. Install the pool lid bolts. See Table 8.1.5 for torque requirements.
  - j. *If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1) Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.*
  - k. *Raise and remove the HI-TRAC from the transfer slide.*
  - ~~k~~.l. Disconnect the MPC slings and lift cleats.
  - ~~l~~.m. Deleted.
  - ~~m~~.n. Deleted.
2. Place HI-TRAC in the designated preparation area.

**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) the HI-TRAC top lid, the top lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

3. Prepare for MPC cool-down as follows:
  - a. Remove the top lid bolts and remove HI-TRAC top lid. See Figure 8.1.9 for rigging.

**Warning:**

At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus flush. Users may also elect the source of water and method for collecting the water flowing from the annulus. Water addition should be performed in a slow and controlled manner until water steam generation has ceased. Water flush should be performed for a minimum of 33 hours at a flow rate of 10 GPM or as specified for the particular heat load of the MPC. *Annulus filling is only required if the SCS is not used.*

- b. *If necessary,* Perform annulus flush by injecting water into the HI-TRAC drain port and allowing the water to cool the MPC shell and lid.
4. *If necessary,* Set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid.
5. Access the MPC as follows:

**ALARA Note:**

The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically vacuumed.

**ALARA Warning:**

Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

- a. Install bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.
    - b. Using the marked locations of the vent and drain ports, core drill the closure ring and vent and drain port cover plates.
6. Remove the closure ring section and the vent and drain port cover plates.

**ALARA Note:**

The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

7. Take an MPC gas sample as follows:

**Note:**

Users may select alternate methods of obtaining a gas sample.

- a. Attach the RVOAs (See Figure 8.1.16).
- b. Attach a sample bottle to the vent port RVOA as shown on Figure 8.3.3.
- c. Using the vacuum drying system, evacuated the RVOA and Sample Bottle.
- d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.
- e. Close the vent port cap and disconnect the sample bottle.

**ALARA Note:**

The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

- f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.
  - g. Remove the drain port cover plate weld and remove the cover plate.
8. Fill the MPC cavity with water as follows:
- a. Configure the cool-down system as shown on Figure 8.3.4.
  - b. Verify that the helium gas pressure regulator is set to the appropriate pressure.
  - c. Open the helium gas supply valve to purge the gas lines of air.
  - d. Deleted.
  - e. If necessary, slowly open the helium supply valve and increase the Cool-Down System pressure. Close the helium supply valve.
  - f. Start the gas coolers.
  - g. Open the vent and drain port caps using the RVOAs.
  - h. Start the blower and monitor the gas exit temperature. Continue the fuel cool-down operations until the gas exit temperature meets the requirements of the Technical Specification LCO 3.1.3.

**Note:**

Water filling should commence immediately at the completion of fuel cool-down operations to prevent fuel assembly heat-up. Prepare the water fill line and the vent line in advance of water filling.

- i. Prepare the MPC fill and vent lines as shown on Figure 8.1.20. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Turn off the blower and disconnect the gas lines to the vent and drain port RVOAs. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

**Note:**

*When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.*

- j. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi (*Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements*). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.
- k. *If used, cease operation of the SCS and remove the system from the HI-TRAC.*

**Caution:**

Oxidation of Boral panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid cutting operations. ~~It is also recommended for defense in depth that the space below the MPC lid be exhausted prior to, and during MPC lid cutting operations to provide additional assurance that explosive gas mixtures will not develop in this space.~~ *The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space*

- ~~k.l.~~ *Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge or evacuate the gas space under the lid as necessary.* ~~Disconnect both lines from the drain and vent ports and, if desired, install an exhaust line to the vent port to evacuate the head space. Perform combustible gas monitoring to ensure there is no combustible mixture present in the exhaust gases.~~
- ~~k.m.~~ Remove the MPC lid-to-shell weld using the weld removal system. See Figure 8.1.9 for rigging.
- ~~k.n.~~ Vacuum the top surfaces of the MPC and HI-TRAC to remove any metal shavings.

9. Install the inflatable annulus seal as follows:

**Caution:**

Do not use any sharp tools or instruments to install the inflatable seal.

- a. Remove the annulus shield.
  - b. Manually insert the inflatable seal around the MPC. See Figure 8.1.13.
  - c. Ensure that the seal is uniformly positioned in the annulus area.
  - d. Inflate the seal
  - e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.
10. Place HI-TRAC in the spent fuel pool as follows:
- a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.
  - b. Engage the lift yoke to HI-TRAC lifting trunnions, remove the MPC lid lifting hole plugs and attach the MPC lid slings or lid retention system to the MPC lid.
  - c. If the lid retention system is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.
  - d. Install the lid retention system bolts if the lid retention system is used.

**ALARA Note:**

The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

- e. If used, fill the annulus overpressure system lines and reservoir with demineralized water and close the reservoir valve. Attach the annulus overpressure system to the HI-TRAC. See Figure 8.1.14.
- f. Position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

**ALARA Note:**

Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

- g. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.
- h. When the top of the HI-TRAC reaches the elevation of the reservoir, open the annulus overpressure system reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.
- i. If the lid retention system is used, remove the lid retention bolts when the top of HI-TRAC is accessible from the operating floor.

- j. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.
- k. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.
- l. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.
- m. Disconnect the drain line from the MPC lid.
- n. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.
- o. Disconnect the lid retention system if used.

#### 8.3.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Vacuum the cells of the MPC to remove any debris or corrosion products.
3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

#### 8.3.5 Post-Unloading Operations

1. Remove HI-TRAC and the unloaded MPC from the spent fuel pool as follows:
  - a. Engage the lift yoke to the top trunnions.
  - b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.
  - c. Raise HI-TRAC until HI-TRAC flange is at the surface of the spent fuel pool.

**ALARA Warning:**  
Activated debris may have settled on the top face of HI-TRAC during fuel unloading.

- d. Measure the dose rates at the top of HI-TRAC in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.
- e. Raise the top of HI-TRAC and MPC to the level of the spent fuel pool deck.
- f. Close the annulus overpressure system reservoir valve.
- g. Using a water pump, lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

**ALARA Note:**  
To reduce contamination of HI-TRAC, the surfaces of HI-TRAC and lift yoke should be kept wet until decontamination can begin.

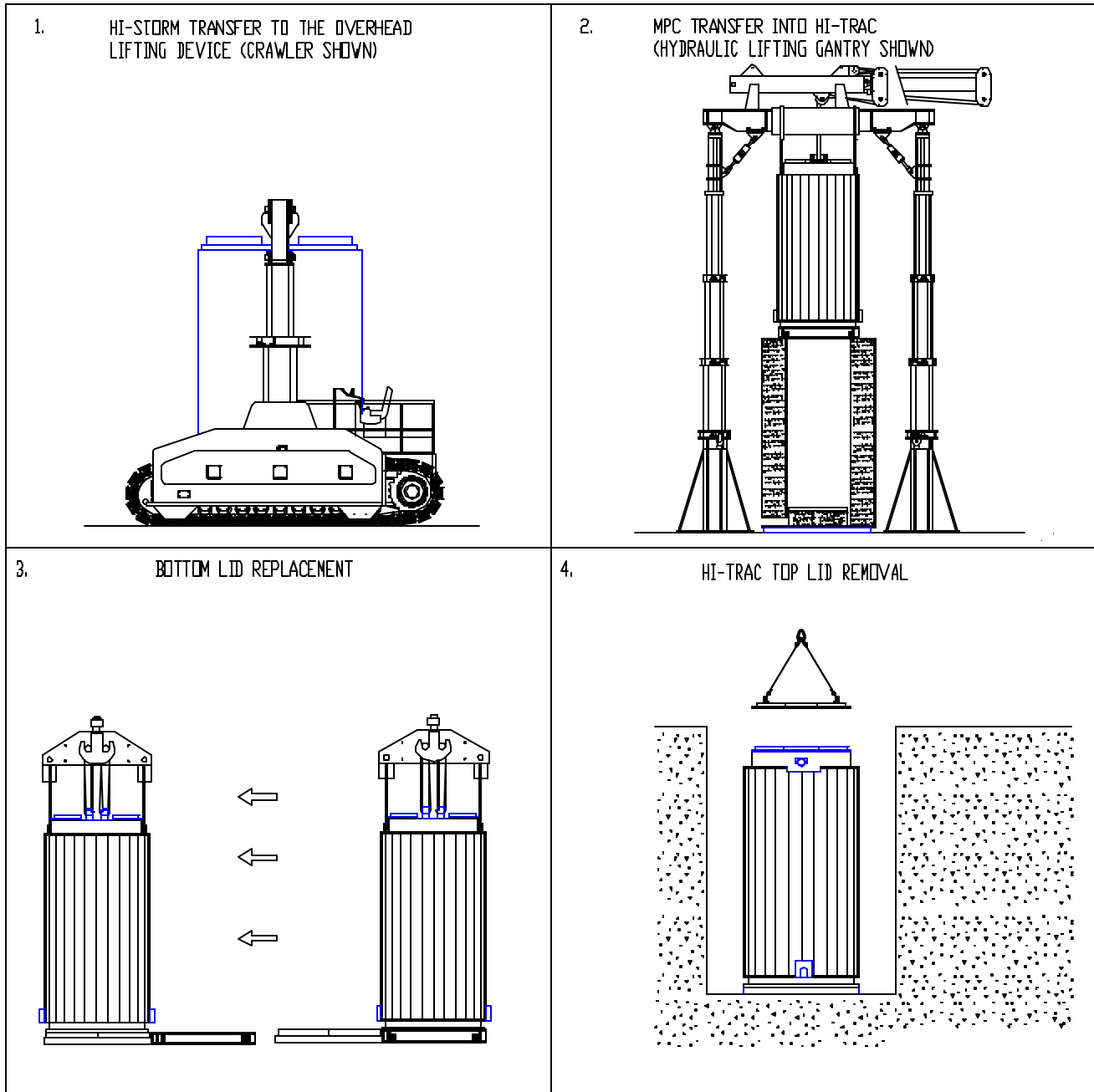
- h. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water.
  - i. Disconnect the annulus overpressure system from the HI-TRAC via the quick disconnect.
  - j. Place HI-TRAC in the designated preparation area.
  - k. Disengage the lift yoke.
  - l. Perform decontamination on HI-TRAC and the lift yoke.
2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.
3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.
4. Drain the water in the annulus area by connecting the drain line to the HI-TRAC drain connector.
5. Remove the MPC from HI-TRAC and decontaminate the MPC as necessary.
6. Decontaminate HI-TRAC.
7. Remove the bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.
8. Return any HI-STORM 100 equipment to storage as necessary.



<b>LOCATION: ISFSI</b>
RECOVER HI-STORM FROM STORAGE
<b>LOCATION: CASK RECEIVING AREA</b>
REMOVE HI-STORM EXIT VENT SCREENS AND CROSS PLATES
REMOVE HI-STORM LID
INSTALL HI-STORM ALIGNMENT DEVICE (OR MATING DEVICE) AND VENT DUCT SHIELD INSERTS
INSTALL MPC LIFT CLEATS
ATTACH MPC SLINGS
INSTALL TOP LID ON HI-TRAC
INSTALL MPC LIFT SLINGS
RAISE HI-TRAC AND MATE OVERPACKS
OPEN SHIELD DOORS (OR MATING DEVICE DRAWER)
ATTACH SLINGS TO LIFT DEVICE AND CLEAT
RAISE MPC INTO HI-TRAC
CLOSE HI-TRAC TRANSFER LID DOORS (HI-TRAC 125D - BOLT UP POOL LID)
REMOVE HI-TRAC FROM TOP OF HI-STORM
DISCONNECT MPC LIFT SLINGS
REPLACE TRANSFER LID WITH POOL LID (NOT FOR HI-TRAC 125D)
DISCONNECT MPC LIFT CLEAT
PLACE HI-TRAC IN DESIGNATED PREPARATION AREA
<b>LOCATION: CASK PREPARATION AREA</b>
REMOVE HI-TRAC TOP LID

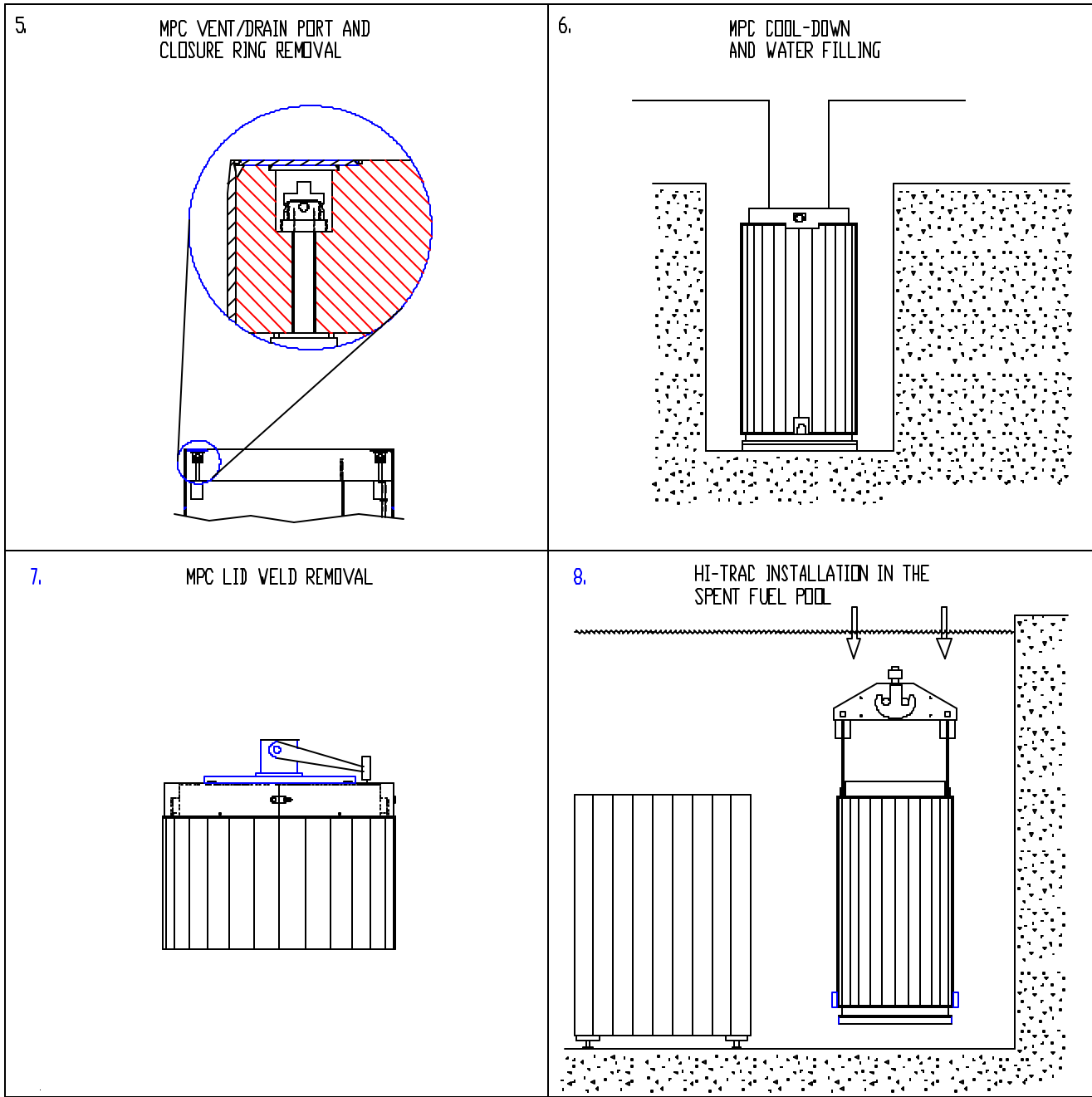
FILL ANNULUS
INSTALL ANNULUS SHIELD
REMOVE MPC CLOSURE RING
REMOVE VENT PORT COVERPLATE WELD AND SAMPLE MPC GAS
PERFORM MPC COOL-DOWN
FILL MPC CAVITY WITH WATER
REMOVE MPC LID TO SHELL WELD
INSTALL INFLATABLE SEAL
PLACE HI-TRAC IN SPENT FUEL POOL
<b>LOCATION: SPENT FUEL POOL</b>
REMOVE MPC LID
DISCONNECT DRAIN LINE
REMOVE SPENT FUEL ASSEMBLIES WASTE FROM MPC
VACUUM CELLS OF MPC
REMOVE HI-TRAC FROM SPENT FUEL POOL
<b>LOCATION: CASK PREPARATION AREA</b>
LOWER WATER LEVEL IN MPC
PUMP REMAINING WATER IN MPC TO SPENT FUEL POOL
REMOVE MPC FROM HI-TRAC
DECONTAMINATE HI-TRAC

**Figure 8.3.1; Unloading Operations Flow Diagram**

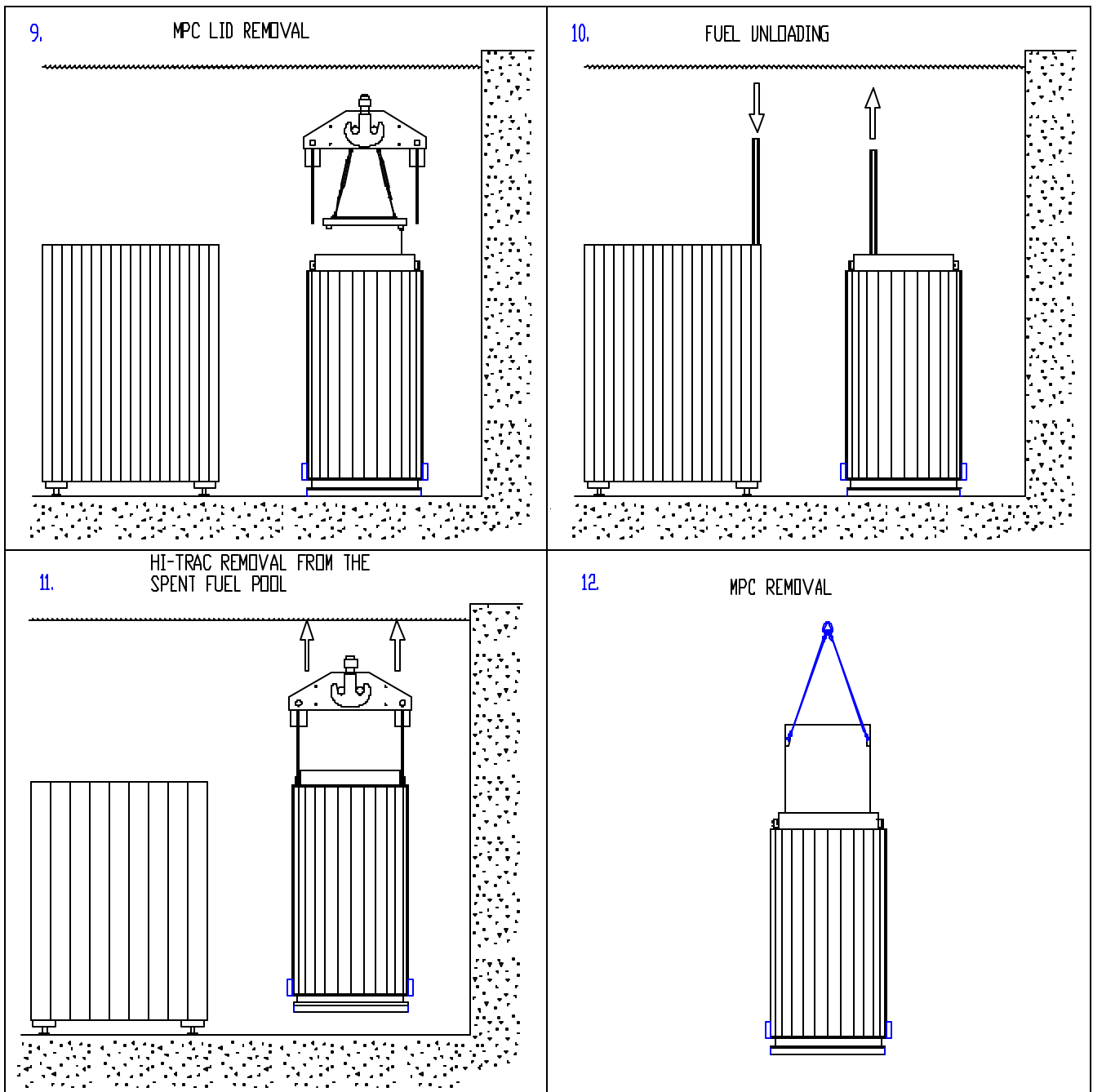


**Note: Bottom Lid Replacement Not Required for HI-TRAC 125D**

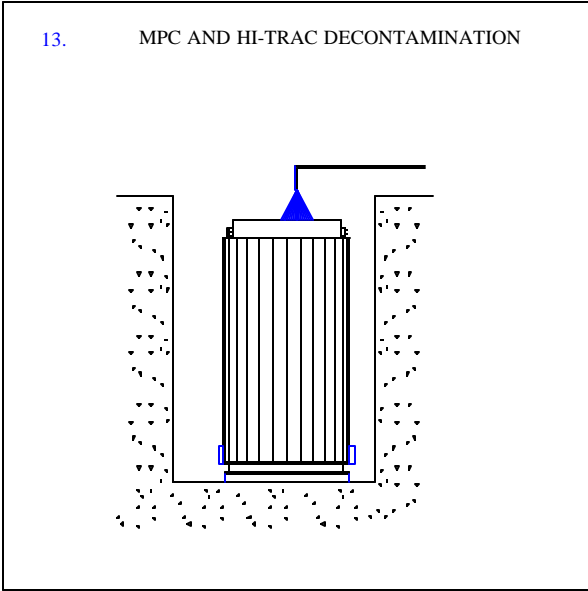
**Figure 8.3.2a; Major HI-STORM 100 Unloading Operations**



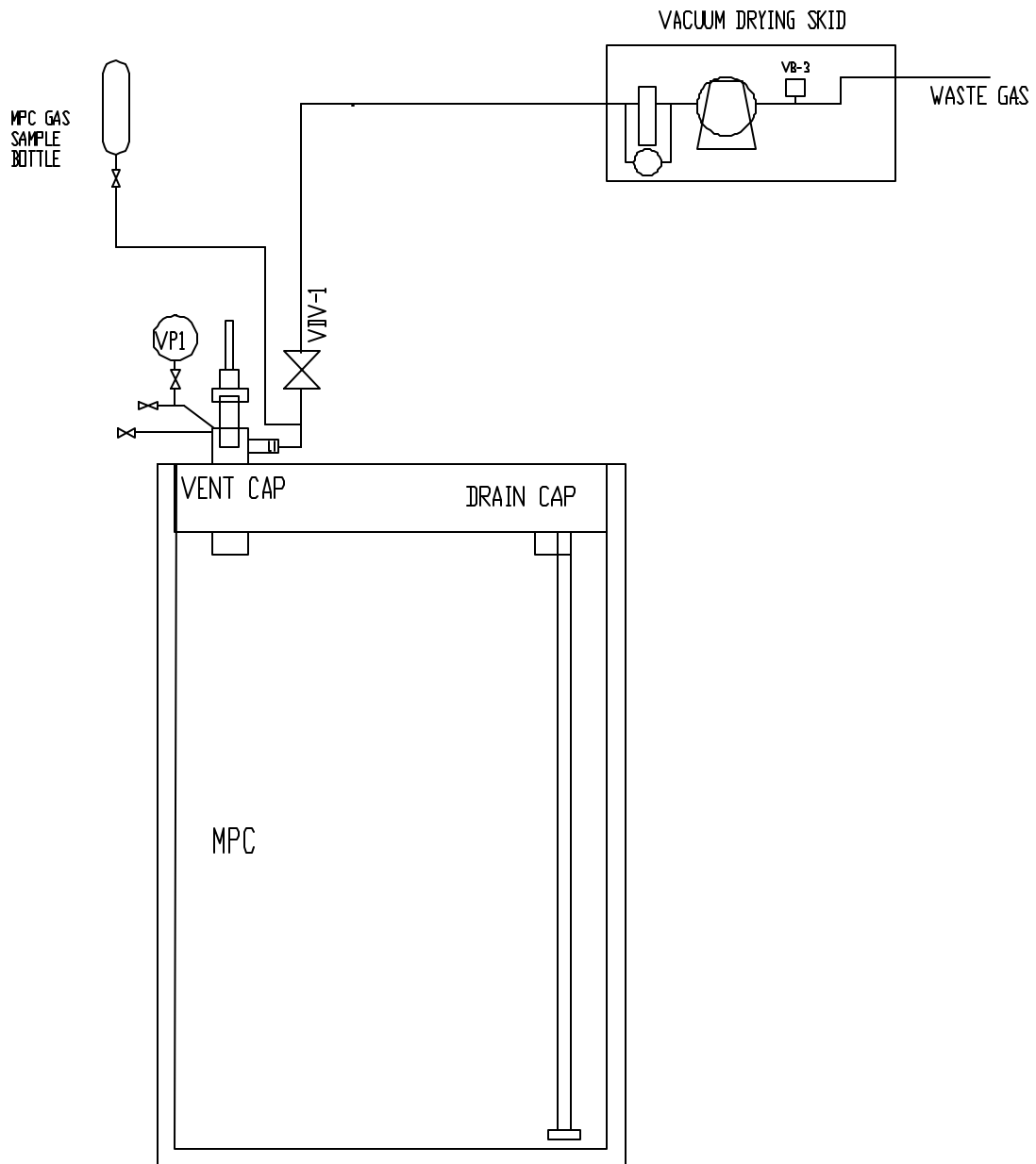
**Figure 8.3.2b; Major HI-STORM 100 Unloading Operations**



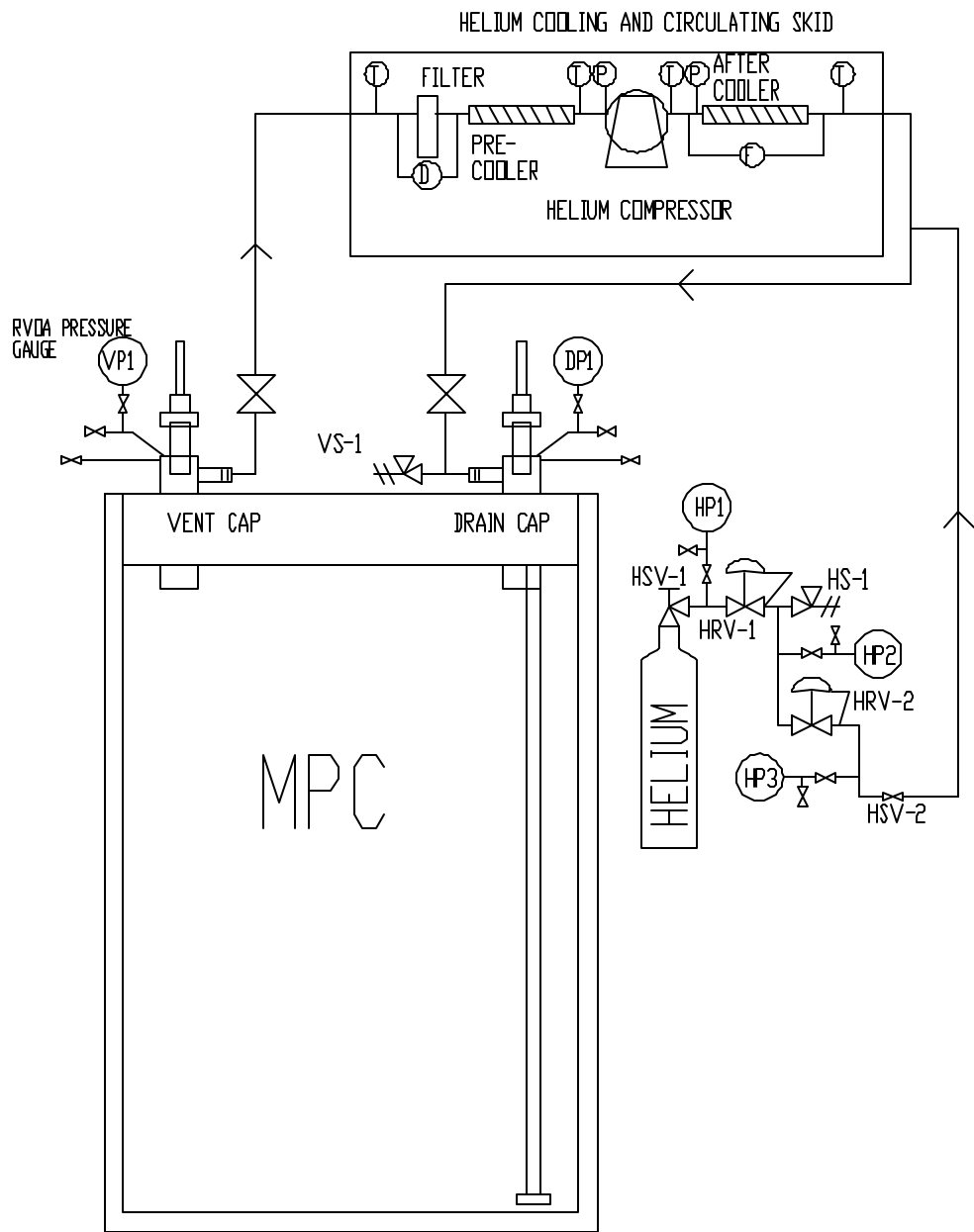
**Figure 8.3.2c; Major HI-STORM 100 Unloading Operations**



**Figure 8.3.2d; Major HI-STORM 100 Unloading Operations**



**Figure 8.3.3; MPC Gas Sampling in Preparation for Unloading**



**Figure 8.3.4; MPC Cool-Down, Example P&ID**

## 8.4 MPC TRANSFER TO A HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE

### 8.4.1 Overview of Operations

The MPC is recovered from storage and transferred into HI-TRAC using the same or similar method as described in Section 8.3. Once the MPC is inside HI-TRAC, the HI-STAR 100 is brought to the transfer location and positioned for receiving of the MPC. If used, the Temporary Shield Ring is installed and filled with water and the Transfer Collar is installed on the HI-STAR 100 Overpack. The Temporary Shield Ring reduces operator dose rates during MPC transfer operations. The Transfer Collar or mating device adapts the top surface of the HI-STAR 100 Overpack to mate with the bottom of HI-TRAC. The MPC may be lowered using the MPC Downloader, the main crane hook or similar device. The MPC slings and MPC lift cleats are attached to the MPC. The MPC is raised slightly, the transfer lid door (or mating device drawer) locking pins are removed and the doors are (drawer is) opened. The MPC is lowered into the HI-STAR. Following verification that the MPC is fully lowered, the MPC slings are disconnected and lowered onto the MPC lid. HI-TRAC is removed from on top of the HI-STAR 100 Overpack. The MPC lift cleat, slings, and the transfer collar/mating device are removed. Hole plugs are installed in the empty MPC lid bolt holes. The HI-STAR 100 Overpack is prepared for storage or transport in accordance with the Certificate of Compliance for storage or transport, as applicable.

### 8.4.2 Recovery from Storage

**Caution:**

*Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.*

1. Recover the MPC from storage and position it inside of HI-TRAC in accordance with Section 8.3.2. ~~See the Technical Specifications for lifting requirements.~~
2. Deleted.

### 8.4.3 MPC Transfer into the HI-STAR 100 Overpack

**Note:**

The following steps outline the HI-STAR 100 operating steps. Refer to the HI-STAR 100 System Final Safety Analysis Report (Docket No. 72-1008) and the HI-STAR 100 System Safety Analysis Report (Docket No. 71-9261) for HI-STAR 100 Overpack specific operations.

1. If necessary, remove the HI-STAR 100 closure plate and the removable shear ring segments. Perform a radiological survey of the inside of the HI-STAR 100 Overpack to verify there is no residual contamination from previous uses. If contamination levels are above specified limits, the HI-STAR 100 Overpack shall be decontaminated appropriately prior to use.



2. Discard any used metallic seals.
3. Perform a HI-STAR 100 receipt inspection in accordance with site-specific procedures.
4. Install the temporary shield ring on HI-STAR 100 and fill it with water, if used. See Figure 8.1.18.
5. Install the HI-STAR transfer collar (or mating device with the pool lid for HI-TRAC 125D). See Figure 8.4.1a (or Figure 8.4.1b).
6. Position HI-STAR adjacent to HI-TRAC.

**Note:**

Lifting of the loaded HI-TRAC shall be performed in accordance with the *applicable lifting requirements*. ~~Technical Specification.~~

7. Raise and align HI-TRAC over HI-STAR and mate the overpacks.

**Note:**

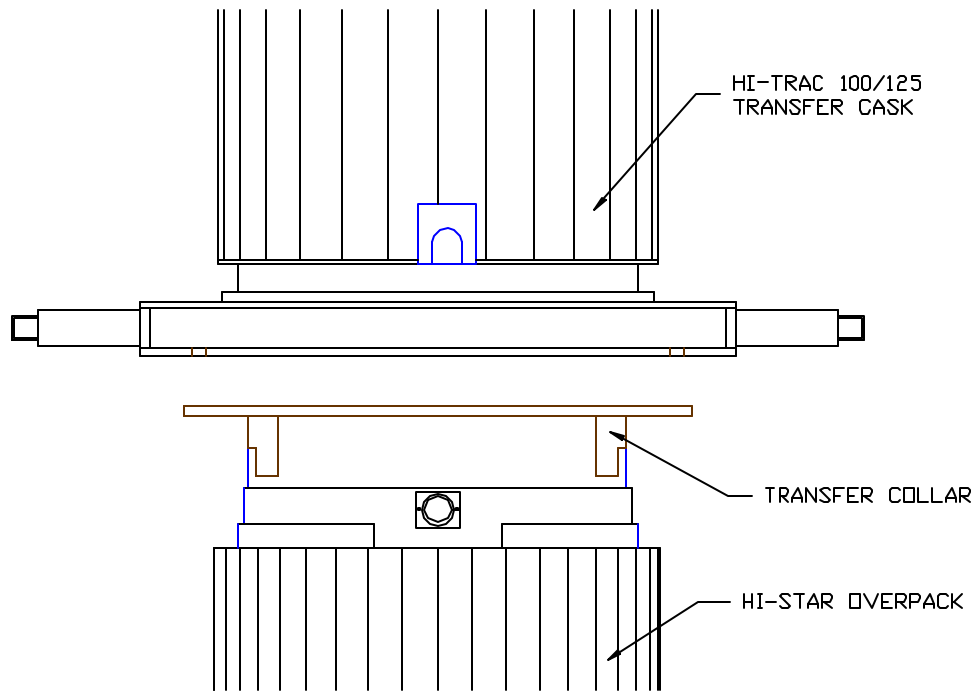
The MPC lift cleats and MPC slings are still installed from the previous operation.

8. Deleted.
9. Remove the transfer lid door (mating device drawer) locking pins and open the doors (drawer).

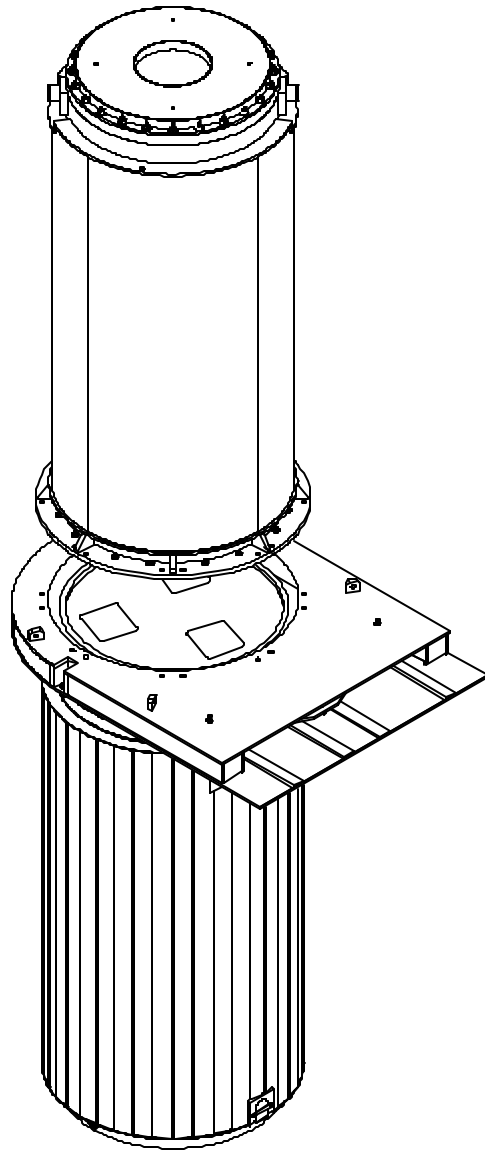
**ALARA Warning:**

If trim plates are not used, personnel should remain clear of the immediate door/drawer area during MPC downloading since there may be radiation streaming during MPC raising and lowering operations.

10. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
11. Lower the MPC into HI-STAR.
12. When the MPC is fully seated, disconnect the slings from the MPC lifting device and lower them on to the MPC lid.
13. Remove HI-TRAC from on top of HI-STAR 100 Overpack.
14. Remove the MPC lift cleat from the MPC and install hole plugs in the empty bolt holes. See Table 8.1.5 for torque requirements.
15. Remove the HI-STAR 100 transfer collar or mating device.
16. Drain and remove the temporary shield ring (if used) and store it in an approved plant storage location.
17. Complete HI-STAR preparation for transport in accordance with the HI-STAR 100 Safety Analysis Report (Docket 71-9261) and the Certificate of Compliance, or complete HI-STAR preparation for storage in accordance with the HI-STAR 100 Final Safety Analysis Report (Docket 72-1008) and the Certificate of Compliance, as applicable.



**Figure 8.4.1a; HI-STAR and HI-TRAC Mating**



**Figure 8.4.1b; HI-STAR and HI-TRAC 125D Mating**

## 8.5 MPC TRANSFER INTO THE HI-STORM 100 OVERPACK DIRECTLY FROM TRANSPORT

### 8.5.1 Overview of Operations

HI-STAR 100 Dual-Purpose Cask System arrives at the receiving location and is surveyed for dose rates and contamination levels. The receiver reviews the shipping paperwork to ensure that the HI-STAR 100 Overpack met the internal contamination limits prior to transportation. The personnel barrier is removed, the impact limiters are removed, the tie-down is removed, and the HI-STAR 100 Overpack is upended. The HI-STAR 100 Overpack is positioned at the designated transfer area and the temporary shield ring is installed. The temporary shield ring reduces operator dose rates during MPC transfer operations. A gas sample is drawn from the annulus and analyzed. The gas sample provides an indication of MPC closure performance. The annulus is depressurized and the closure plate is removed. The transfer collar (mating device with pool lid for HI-TRAC 125D) is installed and the MPC lift cleats are attached to the MPC. The transfer collar (mating device) is used to provide the mating surface on top of the HI-STAR 100 Overpack. The MPC slings are attached to the MPC lift cleat.

If the HI-TRAC 125D is not used, the HI-TRAC is configured with the transfer lid. The top lid<sup>1</sup> is installed, if necessary. HI-TRAC is raised and positioned on top of HI-STAR. The MPC slings are attached to the lifting device. The MPC is raised into HI-TRAC. The HI-TRAC doors/(mating device drawer) are closed and the locking pins are installed. For the HI-TRAC 125D, the pool lid is bolted on. HI-TRAC is raised and the HI-STAR 100 Overpack is removed from under HI-TRAC. The HI-STAR 100 Overpack is repositioned at the user's discretion.

HI-STORM is positioned for MPC receipt with the lid removed, the *alignment device (or mating device) positioned*, and the vent duct shield inserts installed in the exit vent ducts. HI-TRAC is raised and positioned on top of HI-STORM. For HI-TRAC 125D, the pool lid is unbolted. The locking pins are removed and the doors are opened (or mating device drawer opened for HI-TRAC 125D). The MPC is lowered into HI-STORM. The MPC slings are disconnected and lowered onto the MPC lid. HI-TRAC is raised and positioned at the site's discretion. The MPC lift cleat, slings, vent duct shield inserts, and transfer collar (or mating device) are removed and hole plugs are installed in the empty bolt holes. HI-STORM is prepared for storage and transferred to the ISFSI pad in the same manner as described in Section 8.1.

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<sup>1</sup> Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

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## 8.5.2 HI-STAR 100 SYSTEM Receipt and Preparation for MPC Transfer

**Note:**

The following provides a general description of the HI-STAR 100 System operations. Refer to the HI-STAR 100 System Topical Safety Analysis Report (Docket 72-1008) and the Safety Analysis Report (Docket 71-9261) for HI-STAR-specific operations.

1. Review the shipping paperwork and verify that the HI-STAR 100 Overpack met the required internal contamination limits prior to transportation.
2. Measure the HI-STAR 100 dose rates in accordance with 10CFR20.205 [8.5.1].
3. Remove the personnel barrier.
4. Perform removable contamination surveys in accordance with 10CFR20.205 [8.5.1].
5. Remove the impact limiters.
6. Remove the tie-down.
7. Perform a visual inspection of the overpack for obvious signs of shipping damage.
8. Remove the removable shear ring segments from the overpack. (Approximate weight is 50 lbs each).
9. Transfer the HI-STAR 100 Overpack to the location for MPC transfer and position it vertically.
10. Install the temporary shield ring on the overpack top flange if used.

**ALARA Warning:**

Gas sampling is performed to assess the condition of the MPC confinement boundary. If a leak is discovered in the MPC boundary, the MPC may not be placed into HI-STORM. If no leak is detected, the annulus may be vented directly.

11. Perform gas sampling as follows:
  - a. Remove the overpack vent port cover plate and attach the backfill tool with a sample bottle attached. See Figure 8.5.1. Store the cover plate in a site-approved location.
  - b. Using a vacuum pump, evacuate the sample bottle and backfill tool.
  - c. Slowly open the vent port plug and gather a gas sample from the annulus. Reinstall the overpack vent port plug.

- d. Evaluate the gas sample and determine the condition of the MPC confinement boundary.
12. If the confinement boundary is intact (i.e., no radioactive gas is measured) then vent the overpack annulus by removing the overpack vent port seal plug (using the backfill tool). Otherwise return the HI-STAR 100 to the spent fuel pool for MPC unloading in accordance with the HI-STAR 100 SAR.
13. Remove the closure plate bolts and remove the overpack closure plate. Store the closure plate on cribbing to protect the seal seating surfaces. Store the closure plate bolts in a site-approved location.
14. Install the HI-STAR 100 Seal Surface Protector.
15. Install the transfer collar (or mating device with pool lid for HI-TRAC 125D) on HI-STAR. See Figure 8.4.1.

**Note:**

The location of MPC transfer may be selected at the user's discretion.

16. Remove the MPC lift cleat hole plugs and install the MPC lift cleats. See Figure 8.1.24. See Table 8.1.5 for torque requirements.
17. Attach the MPC slings to the MPC lift cleat and lay them on the MPC lid.

**Warning:**

Unless the lift is single-failure proof (or equivalent safety factor) the HI-TRAC top lid, the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

18. If necessary, install the HI-TRAC top lid. See Figure 8.1.9. See Table 8.1.5 for torque requirements.
19. If necessary, configure HI-TRAC with the transfer lid as follows (Not applicable for HI-TRAC 125D):

**ALARA Note:**

The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

- a. Position HI-TRAC vertically adjacent to the transfer lid.
- b. Remove the pool lid bolts and plates and store them in an approved plant storage location.

- c. Raise the empty HI-TRAC and position it on top of the transfer lid.
  - d. Install the bottom lid bolts. See Table 8.1.5 for torque requirements.
20. Position HI-TRAC adjacent to HI-STAR.
  21. Raise HI-TRAC above HI-STAR.
  22. Align HI-TRAC over HI-STAR 100 and mate the overpacks. See Figure 8.1.31.
  23. Remove the locking pins and open the doors or mating device drawer.

**ALARA Warning:**

If trim plates are not being used, personnel should remain clear of the door/drawer area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

24. At the users discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using clamps or any other method deemed suitable by the user. See Figure 8.1.33.

### 8.5.3 Perform MPC Transfer into HI-STORM 100

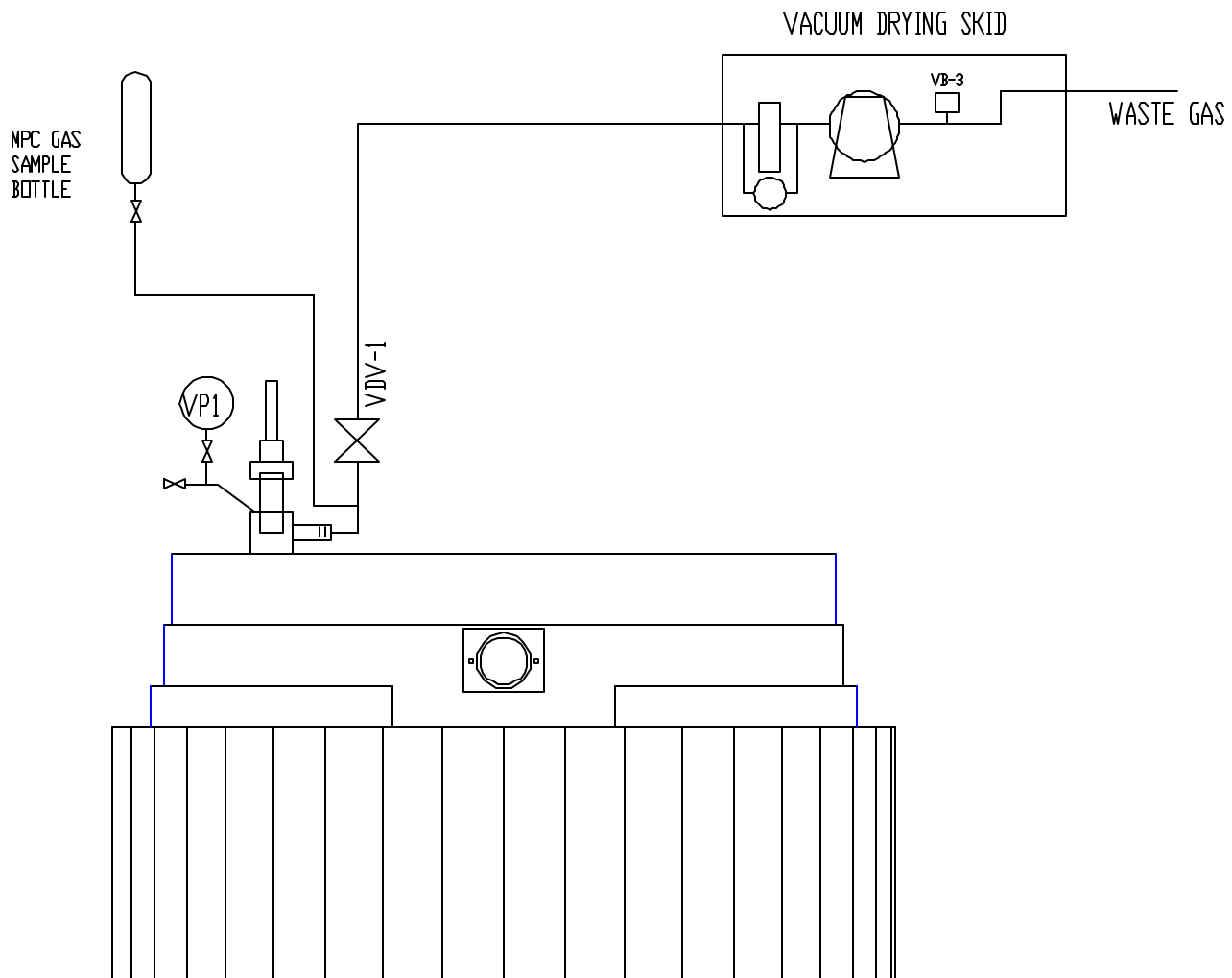
**Caution:**

*Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.*

1. Raise the MPC into HI-TRAC by extending the MPC downloader.
2. Verify the MPC is in the full-up position.
3. Remove the trim plates (if used).
4. Close the HI-TRAC doors/drawer and install the locking pins.
5. For the HI-TRAC 125D, raise the pool lid and bolt it onto the HI-TRAC.
6. Raise HI-TRAC and remove the HI-STAR 100 Overpack from the operations area.
7. Transfer the MPC into HI-STORM in accordance with the steps provided in Section 8.1.
8. Place HI-STORM in storage in accordance with the steps provided in Section 8.1.

9. Perform shielding effectiveness testing ~~in accordance with the HI-STORM 100 System Technical Specification LCO 3.2.3.~~
10. Perform an air temperature rise test per Step 8.1.7.232 if required ~~by the Technical Specifications.~~





**Figure 8.5.1; HI-STAR Annulus Gas Sampling**

## 8.6 REFERENCES

- [8.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.
- [8.1.1] U.S. Code of Federal Regulations, Title 10 "Energy", Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,"
- [8.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.
- [8.1.3] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code".
- [8.5.1] U.S. Code of Federal Regulations, Title 10 " Energy", Part 20, "Standards for Protection Against Radiation,"

## CHAPTER 9<sup>†</sup>: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

### 9.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM 100 System, including the HI-TRAC transfer cask to verify that the structures, systems and components (SSCs) classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM 100 System fabricated, assembled, inspected, tested, and accepted for use under the scope of the HI-STORM 100 System CoC, except as noted herein.

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters ensure that the HI-STORM 100 System will maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions; will maintain subcriticality control; will properly transfer the decay heat of the stored radioactive materials; and that radiation doses will meet regulatory requirements.

Both pre-operational and operational tests and inspections are performed throughout HI-STORM 100 System operations to assure that the HI-STORM 100 System is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, hydrostatic tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 8 identifies the tests and inspections. "Pre-operation", as referred to in this section, defines that period of time from receipt inspection of a HI-STORM 100 System until the empty MPC is loaded into a HI-TRAC transfer cask for fuel assembly loading.

The HI-STORM 100 System is classified as important to safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM 100 System shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance with a quality program commensurate with the particular SSC's graded quality category. Tables 2.2.6 and 8.1.6 provide the quality category for each major item or component of the HI-STORM 100 System and its ancillary equipment, respectively.

The acceptance criteria and maintenance program described in this chapter fully comply with the requirements of 10CFR72 [9.0.1] and NUREG-1536 [9.0.2], except as clarified in Table 1.0.3.

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

## 9.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM 100 System prior to and during first-loading of the system. These inspections and tests provide assurance that the HI-STORM 100 System has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the requirements of 10CFR72 [9.0.1].

~~These inspections and tests are also intended to demonstrate that the operation of the HI-STORM 100 System complies with the applicable regulatory requirements and the Technical Specifications contained in Appendix A to CoC 72-1014. Noncompliances encountered during the required inspections and tests shall be corrected or dispositioned to bring the item into compliance with this FSAR. Identification and resolution of noncompliances shall be performed in accordance with the Holtec International Quality Assurance Program as described in Chapter 13 of this FSAR, or the licensee's NRC-approved Quality Assurance Program.~~

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM 100 overpack, and the 100-ton HI-TRAC and 125-ton HI-TRAC transfer casks are listed in Tables 9.1.1, 9.1.2, and 9.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 8 and 12 provide ~~details on~~ operating *guidance* procedures and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM 100 System has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR.

This section summarizes the test program required for the HI-STORM 100 System.

### 9.1.1 Fabrication and Nondestructive Examination (NDE)

The design, fabrication, inspection, and testing of the HI-STORM 100 System is performed in accordance with the applicable codes and standards specified in Tables 2.2.6 and 2.2.7 and on the Design Drawings. Additional details on specific codes used are provided below.

The following fabrication controls and required inspections shall be performed on the HI-STORM 100 System, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

1. Materials of construction specified for the HI-STORM 100 System are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by ASME Code [9.1.1] Section II (when applicable); the requirements of ASME Section III (when applicable); Holtec procurement specifications; and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to assure material traceability is maintained throughout fabrication. Materials for the confinement boundary (MPC baseplate, lid, closure ring, port cover

plates and shell) shall also be inspected per the requirements of ASME Section III, Article NB-2500.

2. The MPC confinement boundary shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NB, with ~~exceptions~~ *alternatives* as noted below. The MPC basket and basket supports shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NG, with ~~exceptions~~ *alternatives* as noted below. Metal components of the HI-TRAC transfer cask and the HI-STORM overpack, as applicable, shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NF, Class 3 or AWS D1.1, as shown on the design drawings, with ~~exceptions~~ *alternatives* as noted below.

NOTE: ~~Exceptions~~—*NRC-approved alternatives* to these Code requirements are ~~provided~~—*discussed* in FSAR Section 2.2.4, ~~Chapter 2 and in Table 3-1 of Appendix B to CoC 72-1014.~~

3. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable ASME Section III Subsections (e.g., NB, NG, or NF, as applicable to the SSC). AWS code welding may be performed using welders and weld procedures that have been qualified in accordance with applicable AWS requirements or in accordance with ASME Code Section IX
4. Welds shall be visually examined in accordance with ASME Code, Section V, Article 9 with acceptance criteria per ASME Code, Section III, Subsection NF, Article NF-5360, except the MPC fuel basket cell plate-to-cell plate welds and fuel basket support-to-canister welds which shall have acceptance criteria to ASME Code Section III, Subsection NG, Article NG-5360, (as modified by the design drawings). Table 9.1.4 identifies additional nondestructive examination (NDE) requirements to be performed on specific welds, and the applicable codes and acceptance criteria to be used in order to meet the inspection requirements of the applicable ASME Code, Section III. Acceptance criteria for NDE shall be in accordance with the applicable Code for which the item was fabricated. These additional NDE criteria are also specified on the design drawings for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be reviewed and approved by Holtec in accordance with its QA program. NDE inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [9.1.2] or other site-specific, NRC-approved program for personnel qualification.

5. Machined surfaces of the metal components of the HI-STORM 100 System shall be visually examined in accordance with ASME Section V, Article 9, to verify they are free of cracks and pin holes.
6. ASME Code welds requiring weld repair shall be repaired in accordance with the requirements of the ASME Code, Section III, Article NB-4450, NG-4450, or NF-4450, as applicable to the SSC, and examined after repair in the same manner as the original weld.
7. Base metal repairs shall be performed and examined in accordance with the applicable fabrication Code.
8. Grinding and machining operations on the MPC confinement boundary shall be controlled through written and approved procedures and quality assurance oversight to ensure grinding and machining operations do not reduce base metal wall thicknesses of the confinement boundary beyond that allowed per the design drawings. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets Design Drawing requirements. A nonconformance shall be written for areas found to be below allowable base metal thickness and shall be evaluated and repaired per the applicable ASME Code, Subsection NB requirements.
9. Dimensional inspections of the HI-STORM 100 System shall be performed in accordance with written and approved procedures in order to verify compliance to design drawings and fit-up of individual components. All dimensional inspections and functional fit-up tests shall be documented.
10. Required inspections shall be documented. The inspection documentation shall become part of the final quality documentation package.
11. The HI-STORM 100 System shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.
12. Each cask shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

13. A documentation package shall be prepared and maintained during fabrication of each HI-STORM 100 System to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM 100 System or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, but not be limited to:

- Completed Shop Weld Records
- Inspection Records
- Nonconformance Reports
- Material Test Reports
- NDE Reports
- Dimensional Inspection Report

#### 9.1.1.1 MPC Lid-to-Shell Weld Volumetric Inspection

1. The MPC lid-to-shell (LTS) weld shall be volumetrically or multi-layer liquid penetrant (PT) examined following completion of welding. If volumetric examination is used, the ultrasonic testing (UT) method shall be employed. Ultrasonic techniques (including, as appropriate, Time-of-Flight Diffraction, Focussed Phased Array, and conventional pulse-echo) shall be supplemented, as necessary, to ensure substantially complete coverage of the examination volume.
2. If volumetric examination is used, then a PT examination of the root and final pass of the LTS weld shall also be performed and unacceptable indications shall be documented, repaired and re-examined.
3. If volumetric examination is not used, a multi-layer PT examination shall be employed. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed. The 3/8 inch weld depth corresponds to the maximum allowable flaw size determined in Holtec Position Paper DS-213 [9.1.6].
4. It is recognized that welding of the LTS weld may result in indications in the root pass that are not detected by the root pass PT. The overall minimum thickness of the LTS weld has been increased by 0.125 inch such that it is not necessary to take structural credit for the root pass of the weld (actual weld to be a minimum of 0.75 inch). A 0.625-inch J-groove weld was assumed in structural analyses in Chapter 3.
5. For either UT or PT, the maximum detectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection process results,

including *relevant* findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The inspection of the weld shall be performed by qualified personnel and shall meet the acceptance requirements of ASME Section III, NB-5350 for PT and NB-5332 for UT.

6. Evaluation of any indications shall include consideration of any active flaw mechanisms. However, cyclic loading on the LTS weld is not significant, so fatigue is not a factor. The LTS weld is protected from the external environment by the closure ring and the root of the LTS weld is dry and inert (He atmosphere), so stress corrosion cracking is not a concern for the LTS weld.
7. The volumetric or multi-layer PT examination of the LTS weld, in conjunction with other examinations *and tests* performed on this weld (PT of root and final layer, *and hydrostatic-pressure test*), ~~and a helium leakage test~~, the use of ASME Section III acceptance criteria, and the additional weld material added to account for potential defects in the root pass of the weld, in total, provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The volumetric (or multi-layer PT) examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

#### 9.1.2 Structural and Pressure Tests

##### 9.1.2.1 Lifting Trunnions

Two trunnions (located near the top of the HI-TRAC transfer cask) are provided for vertical lifting and handling. The trunnions are designed in accordance with ANSI N14.6 [9.1.3] using a high-strength and high-ductility material (see Chapter 1). The trunnions contain no welded components. The maximum design lifting load of 250,000 pounds for the HI-TRAC 125 and HI-TRAC 125D and 200,000 pounds for the HI-TRAC 100 will occur during the removal of the HI-TRAC from the spent fuel pool after the MPC has been loaded, flooded with water, and the MPC lid is installed. The high-material ductility, absence of materials vulnerable to brittle fracture, large stress margins, and a carefully engineered design to eliminate local stress risers in the highly-stressed regions (during the lift operations) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612 [9.1.4], the acceptance criteria for the lifting trunnions must be established in conjunction with other considerations applicable to heavy load handling.



Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure (including trunnion failure) is one:

- i. operator errors
- ii. rigging failure
- iii. lack of adequate inspection
- iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load (750,000 lbs for the HI-TRAC 125 and HI-TRAC 125D and 600,000 lbs for the HI-TRAC 100) shall be applied for a minimum of 10 minutes. The accessible parts of the trunnions (areas outside the HI-TRAC cask), and the adjacent HI-TRAC cask trunnion attachment area shall then be visually examined to verify no deformation, distortion, or cracking occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-TRAC cask trunnion attachment areas shall require replacement of the trunnion and/or repair of the HI-TRAC cask. Following any replacements and/or repair, the load testing shall be performed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing shall be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section II requirements will provide further verification of the trunnion load capabilities. Test results shall be documented. The documentation shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above will provide adequate assurance against handling accidents.

#### 9.1.2.2 Hydrostatic Pressure Testing

##### 9.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

The 125-ton (including HI-TRAC 125 and HI-TRAC 125D) and 100-ton HI-TRAC transfer cask water jackets shall be hydrostatically tested to 75 psig +3,-0 psig, and 71 psig +3, -0 psig, respectively, in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten

minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Liquid penetrant (PT) or magnetic particle (MT) examination of accessible welds shall be performed in accordance with ASME Code, Section V, Articles 6 and 7, respectively, with acceptance criteria per ASME Code, Section III, Subsection NF, Articles NF-5350 and NF-5340, respectively. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

If a hydrostatic retest is required and fails, a nonconformance report shall be issued and a root cause evaluation and appropriate corrective actions taken before further repairs and retests are performed.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

#### 9.1.2.2.2 MPC Confinement Boundary

~~Hydrostatic~~ Pressure testing (*hydrostatic or pneumatic*) of the MPC confinement boundary shall be performed in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 *and applicable sub-articles*, when field welding of the MPC lid-to-shell weld is completed. *If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure.* ~~The hydrostatic pressure for the test is 125 +5, 0 psig, which is 125% of the design pressure of 100 psig.~~ The MPC vent and drain ports will be used for pressurizing the MPC cavity. The loading procedures in FSAR Chapter 8 define the test equipment arrangement. The calibrated test pressure gage installed on the MPC confinement boundary shall have an upper limit of approximately twice that of the test pressure. Following completion of the ~~10-minute~~ *required* hold period at the ~~hydrostatic~~ test pressure, ~~and while maintaining a minimum test pressure of 125 psig,~~ the surface of the MPC lid-to-shell weld shall be ~~visually examined for leakage and then~~ re-examined by liquid penetrant examination in accordance with ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. The performance and sequence of the test is described in FSAR Section 8.1 (loading procedures).

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, *if applicable*, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, *Article NB-4450*. ~~Subsection NB, NB-4450.~~

The MPC confinement boundary ~~hydrostatic pressure~~ test shall be repeated until all ~~visual and liquid penetrant~~ *required* examinations are found to be acceptable. ~~in accordance with the acceptance criteria.~~ Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

#### 9.1.2.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested include the components identified in Table 3.1.18 and applicable weld materials. Table 3.1.18 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as described in FSAR Appendix 1.D in accordance with written and approved procedures. Testing shall verify the composition, compressive strength, and density meet design requirements.

Concrete testing shall be performed for each lot of concrete. Concrete testing shall comply with ACI 349, as described in Table 1.D.2. Test specimens shall be in accordance with ASTM C39.

Test results shall be documented and become part of the final quality documentation package.

#### 9.1.3 Leakage Testing

Leakage testing shall be performed in accordance with the requirements of ANSI N14.5 [9.1.5]. Testing shall be performed in accordance with written and approved procedures.

At completion of welding the MPC shell to the baseplate, an MPC confinement boundary weld helium leakage test shall be performed using a helium mass spectrometer leak detector (MSLD). A temporary test closure lid is used in order to provide a sealed MPC. The confinement boundary welds shall have indicated helium leakage rates less than or equal to  $5 \times 10^{-6}$  atm cm<sup>3</sup>/s (helium). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criteria is met.

If failure of the leakage rate retest occurs after initial repairs are completed, a nonconformance report shall be issued, and a root cause evaluation and appropriate corrective actions taken before further repairs and retest are performed.

Leakage testing of the field-welded MPC lid-to-shell weld, *vent and drain port cover plate welds, and closure ring welds is not required.* ~~shall be performed following the successful completion of~~

~~the MPC hydrostatic test performed per Section 9.1.2.2.2. Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in FSAR Section 8.1, and the acceptance criteria are defined in the Technical Specifications in Appendix A to CoC 72-1014.~~

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

#### 9.1.4 Component Tests

##### 9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no fluid transport devices or rupture discs associated with the HI-STORM 100 System. The only valve-like components in the HI-STORM 100 System are the specially designed caps installed in the MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully-welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and are ~~leak tested~~ *liquid penetrant examined* to verify the MPC confinement boundary.

There are two pressure relief valves installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. These pressure relief valves are ~~is provided~~ for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The pressure relief valves shall relieve at 60 psig and 65 psig.

##### 9.1.4.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM 100 System.

#### 9.1.5 Shielding Integrity

The HI-STORM overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM overpack concrete provides both neutron and gamma shielding. Additional neutron shielding is provided by the encased ~~Boral~~ neutron absorber attached to the fuel basket cell surfaces inside the MPCs. The overpack's inner and outer steel shells, and the steel shield shell<sup>†</sup>, provide radial gamma shielding. Concrete and steel plates provide axial neutron and gamma shielding. A concrete ring attached to the top of the overpack lid provides additional gamma and neutron shielding in the axial direction. Steel gamma shield cross plates, installed in the overpack air inlet and outlet vents, provide additional shielding for radiation through the vent openings.

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<sup>†</sup> The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated. Those overpacks without the shield shell are required to have a higher concrete density in the overpack body to provide compensatory shielding. See Table 1.D.1.

The HI-TRAC transfer cask uses three different materials for primary shielding. All three HI-TRAC transfer cask designs include a radial steel-lead-steel shield and a steel-lead-steel pool lid design. The top lid in the HI-TRAC 125 and HI-TRAC 125D designs includes Holtite neutron shielding inside a steel enclosure. The HI-TRAC 100 top lid includes only steel shielding. The HI-TRAC 125 transfer lid includes steel, lead, and Holtite, while the HI-TRAC 100 includes only steel and lead. The HI-TRAC 125D design does not include a transfer lid. The water jacket, included in all transfer cask designs, provides radial neutron shielding. Testing requirements for the shielding items are described below.

#### 9.1.5.1 Fabrication Testing and Control

##### Holtite-A:

Neutron shield properties of Holtite-A are provided in Chapter 1, Section 1.2.1.3.2. Each manufactured lot of neutron shield material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration and neutron shield density (or specific gravity) meet the requirements specified in Chapter 1 and the Bill-of-Material. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package.

The installation of the neutron shielding material shall be performed in accordance with written and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

##### Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR Appendix 1.D and the design drawings prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 9.1.2.3 for concrete material testing requirements.

##### Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure voids are minimized. Stand pipes or similar devices (as applicable) shall be placed on the upper portion of the cask to ensure the presence of an excess head of liquid lead. Vent risers shall be provided to allow for the air to escape. The HI-TRAC cask components shall be uniformly preheated prior to a lead pour. The temperature of the lead shall be verified to be in the correct temperature range and the lead shall be poured or pumped into place in

the annulus (as applicable). The lead pour shall be followed by a controlled cooldown to minimize the gap between the lead and steel shells. The lead shall be cooled from the bottom up and additional molten lead shall be added to the standpipes as necessary to account for lead shrinkage. Each lot of lead shall be tested for chemical composition.

As an alternative to pouring molten lead, the HI-TRAC lead shielding may be installed as pre-cast sections. If pre-cast sections are used, the design of the sections and the installations instructions shall minimize the gaps between adjacent lead sections and between the lead and the transfer cask walls to the extent practicable.

Steel:

Steel plates utilized in the construction of the HI-STORM 100 System shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

General Requirements for Shield Materials:

1. Test results shall be documented and become part of the quality documentation package.
2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed during fabrication and again after initial loading operations in accordance with Section 9.1.5.2 below and the operating procedures in Chapter 8.

#### 9.1.5.2 Shielding Effectiveness Tests

The effectiveness of the lead pours in the HI-TRAC transfer cask body shall be verified during fabrication by performing gamma scanning on all accessible surfaces of the cask in the lead pour region. The gamma scanning may be performed prior to, or after installation of the water jacket. The purpose of the gamma scanning test is to demonstrate that the gamma shielding of the transfer cask body is at least as effective as that of a lead and steel test block. For the test block, the steel thickness shall be equivalent to the minimum design thickness of steel in the transfer cask component and the lead thickness shall be 5 percent lower than the minimum design thickness of lead in the transfer cask component (see the Design Drawings for the design values). Data shall be recorded on a 6-inch by 6-inch (nominal) grid pattern over the surfaces to be scanned. Should the measured gamma dose rates exceed those established with the test block, the shielding of that transfer cask component shall be deemed unacceptable. Corrective actions should be taken, if practicable, and the testing re-performed until successful results are achieved. If physical corrective actions are not practicable, the degraded condition may be dispositioned with a written evaluation in accordance with applicable procedures to determine the acceptability of the transfer cask for service. Gamma scanning shall be performed in accordance with written and approved procedures. Dose rate measurements shall be documented and shall become part of the quality documentation package.

The effectiveness of the lead plates in the HI-TRAC pool lid (all transfer cask designs) and transfer lid (HI-TRAC 125 and 100 only) shall be verified during fabrication by performing a UT test of the lead plates. The UT testing will take place before the installation of the plates. The UT testing ensures that the plates are uniform internally. This is an accepted industry procedure for locating voids within the lead plate in order to verify the shielding effectiveness of the plate.

Following the first fuel loading of each HI-STORM 100 System (HI-TRAC transfer cask and HI-STORM storage overpack), a shielding effectiveness test shall be performed at the loading facility site to verify the effectiveness of the radiation shield. This test shall be performed after the HI-STORM overpack and HI-TRAC transfer cask have been loaded with an MPC containing spent fuel assemblies and the MPC has been drained, moisture removed, and backfilled with helium.

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM overpack and HI-TRAC. Measurements shall be taken at the locations specified in the ~~technical specifications in Appendix A to CoC 72-1014 and, if necessary, average dose rates computed~~ *Radiation Protection Program* for comparison against the prescribed limits. ~~The results of the dose rate measurements shall be compared to the limits specified in the technical specifications.~~ The test is considered acceptable if the dose rate readings are less than or equal to *the calculated* limits. ~~in the technical specifications.~~ If dose rates are higher than the limits, the required actions provided in the ~~technical specifications~~ *Radiation Protection Program* shall be completed. Dose rate measurements shall be documented and shall become part of the quality documentation package.

*NOTE*

*Section 9.1.5.3 below (including Subsections 9.1.5.3.1 through 9.1.5.3.3) is incorporated into the HI-STORM 100 CoC by reference (CoC Appendix B, Section 3.2.8) and may not be deleted or altered in any way without prior NRC approval via CoC amendment. The text of this section is, therefore, shown in bold type to distinguish it from other text.*

### **9.1.5.3      Neutron Absorber Tests**

**Each plate of ~~Boral~~ neutron absorber shall be visually inspected by the manufacturer for damage such as (e.g., scratches, cracks, burrs, and peeled cladding,) and foreign material embedded in the surfaces. In addition, the MPC fabricator shall visually inspect the Boral plates on a lot sampling basis. The sample size shall be determined in accordance with MIL-STD-105D or equivalent. The selected ~~neutron absorber~~ Boral plates shall be inspected for damage such as inclusions, cracks, voids, delamination, and surface finish, as applicable.**

#### 9.1.5.3.1 *Boral* (75% Credit)

After manufacturing, a statistical sample of each lot of *neutron absorber Boral* shall be tested using wet chemistry and/or neutron attenuation *testing techniques* to verify a minimum  $^{10}\text{B}$  content (areal density) ~~at~~ *in samples taken from* the ends of the panel. The minimum  $^{10}\text{B}$  loading of the *neutron absorber Boral* panels for each MPC model is provided in Table 2.1.15. Any panel in which  $^{10}\text{B}$  loading is less than the minimum allowed shall be rejected. Testing shall be performed using written and approved procedures. Results shall be documented and become part of the cask quality records documentation package.

#### 9.1.5.3.2 *METAMIC*<sup>®</sup> (90% Credit)

*NUREG/CR-5661 identifies the main reason for a penalty in the neutron absorber B-10 density as the potential of neutron streaming due to non-uniformities in the neutron absorber, and recommends comprehensive acceptance tests to verify the presence and uniformity of the neutron absorber for credits more than 75%. Since a 90% credit is taken for METAMIC<sup>®</sup>, the following criteria must be satisfied:*

- *The boron carbide powder used in the manufacturing of METAMIC<sup>®</sup> must have small particle sizes to preclude neutron streaming*
- *The  $^{10}\text{B}$  areal density must comply with the limits of Table 2.1.15.*
- *The  $\text{B}_4\text{C}$  powder must be uniformly dispersed locally, i.e. must not show any particle agglomeration. This precludes neutron streaming.*
- *The  $\text{B}_4\text{C}$  powder must be uniformly dispersed macroscopically, i.e. must have a consistent concentration throughout the entire neutron absorber panel.*
- *The maximum  $\text{B}_4\text{C}$  content in METAMIC<sup>®</sup> shall be less than or equal to 33.0 weight percent.*

*To ensure that the above requirements are met the following tests shall be performed:*

- *All lots of boron carbide powder are analyzed to meet particle size distribution requirements.*
- *The following qualification testing shall be performed on the first production run of METAMIC<sup>®</sup> panels for the MPCs in order to validate the acceptability and consistency of the manufacturing process and verify the acceptability of the METAMIC<sup>®</sup> panels for neutron absorbing capabilities:*
  - 1) *The boron carbide powder weight percent shall be verified by testing a sample from forty different mixed batches. (A mixed batch is defined as a single mixture of aluminum powder and boron carbide powder used to make one or more billets. Each billet will produce several panels.) The samples shall be drawn from the mixing containers after*



*mixing operations have been completed. Testing shall be performed using the wet chemistry method.*

- 2) The  $^{10}\text{B}$  areal density shall be verified by testing a sample from one panel from each of forty different mixed batches. The samples shall be drawn from areas contiguous to the manufactured panels of METAMIC<sup>®</sup> and shall be tested using the wet chemistry method. Alternatively, or in addition to the wet chemistry tests, neutron attenuation tests on the samples may be performed to quantify the actual  $^{10}\text{B}$  areal density.*
  - 3) To verify the local uniformity of the boron particle dispersal, neutron attenuation measurements of random test coupons shall be performed. These test coupons may come from the production run or from pre-production trial runs.*
  - 4) To verify the macroscopic uniformity of the boron particle distribution, test samples shall be taken from the sides of one panel from five different mixed batches before the panels are cut to their final sizes. The sample locations shall be chosen to be representative of the final product. Wet chemistry or neutron attenuation shall be performed on each of the samples.*
- During production runs, testing of mixed batches shall be performed on a statistical basis to verify the correct boron carbide weight percent is being mixed.*
  - During production runs, samples from random METAMIC<sup>®</sup> panels taken from areas contiguous to the manufactured panels shall be tested via wet chemistry to verify the  $^{10}\text{B}$  areal density. This test shall be performed to verify the continued acceptability of the manufacturing process.*

*The measurements of  $\text{B}_4\text{C}$  particle size,  $^{10}\text{B}$  isotopic assay, uniformity of  $\text{B}_4\text{C}$  distribution and  $^{10}\text{B}$  areal density shall be made using written and approved procedures. Results shall be documented.*

#### **9.1.5.3.3 Installation of the Neutron Absorber Panels**

Installation of ~~neutron absorber~~ **Boral** panels into the fuel basket shall be performed in accordance with written and approved instructions. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a *neutron absorber* ~~Boral~~ panel in accordance with the drawings in Chapter 1. These quality control processes, in conjunction with ~~Boral~~ *in-process* manufacturing testing, provide the necessary assurances that the *neutron absorber* ~~Boral~~ will perform its intended function. No additional testing or in-service monitoring of the *neutron absorber material* ~~Boral~~ will be required.

#### **9.1.6 Thermal Acceptance Tests**

The thermal performance of the HI-STORM 100 System, including the MPCs and HI-TRAC transfer casks, is demonstrated through analysis in Chapter 4 of the FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed

prior to system loading. Following the loading and placement on the storage pad of the first HI-STORM System placed in service, the operability of the natural convective cooling of the HI-STORM 100 System shall be verified by the performance of an air temperature rise test. A description of the test is described in FSAR Chapter 8.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM 100 heat removal system.

#### 9.1.7 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in the Design Drawings in Chapter 1.

Table 9.1.1  
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<ul style="list-style-type: none"> <li>a) Examination of MPC components per ASME Code Section III, Subsections NB and NG, as defined on design drawings, per NB-5300 and NG-5300, as applicable.</li> <li>b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements.</li> <li>c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements.</li> <li>d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations.</li> <li>e) Cleanliness of the MPC shall be verified upon completion of fabrication.</li> <li>f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.</li> </ul>	<ul style="list-style-type: none"> <li>a) The MPC shall be visually inspected prior to placement in service at the licensee's facility.</li> <li>b) MPC protection at the licensee's facility shall be verified.</li> <li>c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.</li> </ul>	<ul style="list-style-type: none"> <li>a) None.</li> </ul>

Table 9.1.1 (continued)  
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	<p>a) Assembly and welding of MPC components shall be performed per ASME Code Section IX and III, Subsections NB and NG, as applicable.</p> <p>b) Materials analysis (steel, <i>neutron absorber</i><del>Boral</del>, etc.), shall be performed and records shall be kept in a manner commensurate with "important to safety" classifications.</p>	<p>a) None.</p>	<p>a) An ultrasonic (UT) examination or multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld shall be performed per ASME Section V, Article 5 (or ASME Section V, Article 2). Acceptance criteria for the examination are defined in Subsection 9.1.1.1 and in the Design Drawings.</p> <p>b) ASME Code NB-6000 <del>hydrostatic pressure</del> test shall be performed after MPC closure welding. Acceptance criteria are defined in <del>Subsection 9.1.2.2.2</del><i>the Code</i>.</p>
Leak Tests	<p>a) Helium leak rate testing shall be performed on all MPC pressure boundary shop welds.</p>	<p>a) None.</p>	<p>a) <del>None. Helium leak rate testing shall be performed on MPC lid to shell, and vent and drain ports to MPC lid field welds after closure welding. Acceptance criteria are defined in the technical specifications.</del></p>

Table 9.1.1 (continued)  
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Criticality Safety	a) The boron content shall be verified at the time of neutron absorber material manufacture.	a) None.	a) None.
	b) The installation of <i>neutron absorber</i> <del>Boral</del> panels into MPC basket plates shall be verified by inspection.		
Shielding Integrity	a) Material compliance shall be verified through CMTRs.	a) None.	a) None.
	b) Dimensional verification of MPC lid thickness shall be performed.		
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is to be tested during fabrication.  - MPC lid - vent/drain port cover plates - MPC closure ring	a) Fit-up of the following components shall be verified during pre-operation.  - MPC lid - MPC closure ring - vent/drain cover plates	a) None.
	b) A gauge test of all basket fuel compartments.		
Canister Identification Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 9.1.2

HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
<p>Visual Inspection and Nondestructive Examination (NDE)</p>	<p>Structural Steel Components:</p> <ul style="list-style-type: none"> <li>a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</li> <li>b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</li> <li>c) All welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</li> <li>d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations.</li> </ul> <p>Concrete Components: The following processes related to concrete components shall be implemented per ACI 349 as clarified in FSAR Appendix 1.D. Concrete testing shall be in accordance with Table 1.D.2. Activities shall be conducted in accordance with written and approved procedures.</p> <ul style="list-style-type: none"> <li>a) Assembly and examination.</li> <li>b) Materials verification.</li> <li>c) Mixing, pouring, and testing.</li> </ul>	<ul style="list-style-type: none"> <li>a) The overpack shall be visually inspected prior to placement in service.</li> <li>b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means.</li> <li>c) Overpack protection at the licensee's facility shall be verified.</li> <li>d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.</li> </ul>	<ul style="list-style-type: none"> <li>a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned .</li> <li>b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.</li> </ul>

Table 9.1.2 (continued)  
HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) No structural or pressure tests are required for the overpack during fabrication. b) Concrete compressive strength tests shall be performed per ASTM C39.	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per ACI-349 as clarified by FSAR Appendix 1.D, at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement. c) Verification of material composition shall be performed.	a) None	a) A shielding effectiveness <i>test</i> shall be performed after the initial fuel loading. <del>in accordance with the Technical Specifications.</del> Repeat shielding effectiveness test every five years as part of the Maintenance Program described in FSAR Section 9.2.

Table 9.1.2 (continued)  
 HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM 100 System in accordance with the operating procedures in Chapter 8.  b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's discretion. <del>in accordance with the technical specifications:</del>  (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions.  (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading..	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.



Table 9.1.3  
HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<ul style="list-style-type: none"> <li>a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</li> <li>b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</li> <li>c) All welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</li> <li>d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations</li> <li>e) Cleanliness of the transfer cask shall be verified upon completion of fabrication.</li> <li>f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.</li> </ul>	<ul style="list-style-type: none"> <li>a) The transfer cask shall be visually inspected prior to placement in service.</li> <li>b) Transfer cask protection at the licensee's facility shall be verified.</li> <li>c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.</li> </ul>	<ul style="list-style-type: none"> <li>a) Annual visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements. (See footnote for Table 9.2.1).</li> </ul>

Table 9.1.3 (continued)  
 HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category.  a) A load test of the lifting trunnions shall be performed during fabrication per ANSI N14.6. b) A pressure test of the neutron shield water jacket shall be performed during fabrication.	a) None.	a) Annual load testing of the lifting trunnions shall be performed per ANSI N14.6. (See footnote to Table 9.2.1).  b) The set pressure of the relief valve on the neutron shield water jacket shall be verified by calibration annually. (See footnote to Table 9.2.1)
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) None.	a) None.	a) None.

Table 9.1.3 (continued)  
TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading..	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask components (top, in-pool, and transfer lids) shall be performed during fabrication.	a) Fit-up test of the transfer cask lifting trunnions with the transfer cask lifting yoke shall be performed.  b) Fit-up test of the transfer cask pocket trunnions with the horizontal transfer skid shall be performed.	a) Fit-up of the top, in-pool, and transfer lids shall be verified prior to use.

Table 9.1.4  
HI-STORM 100 NDE REQUIREMENTS

<b>MPC</b>			
<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
Shell longitudinal seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT or UT	ASME Section V, Article 2 (RT) ASME Section V, Article 5 (UT)	RT: ASME Section III, Subsection NB, Article NB-5320 UT: ASME Section III, Subsection NB, Article NB-5330
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350

Table 9.1.4 (continued)  
HI-STORM 100 NDE REQUIREMENTS

<b>MPC</b>			
<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
Lid-to-shell	PT (root and final pass) and multi-layer PT (if UT is not performed).	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
	PT (surface following hydrostatic pressure test) UT (if multi-layer PT is not performed)	ASME Section V, Article 5 (UT)	UT: ASME Section III, Subsection NB, Article NB-5332
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug, lift lug baseplate, and fuel spacers	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350

Table 9.1.4 (continued)  
 HI-STORM 100 NDE REQUIREMENTS

<b>HI-STORM OVERPACK</b>			
<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
N/A	N/A	N/A	N/A
<b>HI-TRAC TRANSFER CASK</b>			
<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
HI-TRAC Body: Radial ribs and short ribs to outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Water jacket end plate-to-radial channel or enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid top plate-to-pool lid outer ring [HI-TRAC 125 and 125D only]	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid bottom plate-to-pool lid outer ring [HI-TRAC 125 and 125D only]	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued)  
 HI-STORM 100 NDE REQUIREMENTS

**HI-TRAC TRANSFER CASK**

<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
HI-TRAC Body: Water jacket end plate-to-outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-outer shell longitudinal and circumferential welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Radial ribs and short ribs –to-enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Jacket drain pipe and couplings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-bottom flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued)  
 HI-STORM 100 NDE REQUIREMENTS

**HI-TRAC TRANSFER CASK**

<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
HI-TRAC Body: Outer shell-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-outer and inner shells	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-outer shell (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Top lid welds except as noted on applicable drawings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-enclosure shell panel and radial rib (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340



Table 9.1.4 (continued)  
 HI-STORM 100 NDE REQUIREMENTS

<b>HI-TRAC TRANSFER CASK</b>			
<b>Weld Location</b>	<b>NDE Requirement</b>	<b>Applicable Code</b>	<b>Acceptance Criteria (Applicable Code)</b>
HI-TRAC Body: Lower water jacket welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Gusset-to-baseplate, outer shell and water jacket bottom plate (HI-TRAC 125D only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lid intermediate plate and lead cover plate-to-lid top plate & lid bottom plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door top plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door end plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lead cover plate-to-lead cover side plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

## 9.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM 100 System Operations Manual which shall be prepared and issued prior to the delivery and first use of the system to each user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal, and confinement performance; radiological safety, and proper handling of the system in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM 100 System is totally passive by design. There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from weathering effects in storage. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures no more demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STORM 100 System is provided in Table 9.2.1.

### 9.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC lifting trunnions and pocket trunnion recesses. The examination shall inspect for indications of overstress such as cracking, deformation, or wear marks. Repairs or replacement in accordance with written and approved procedures shall be required if unacceptable conditions are identified.

A load test on the transfer cask trunnions shall be performed annually or prior to the next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year. The requirements are specified in Section 9.1.2.1.

As described in FSAR Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

### 9.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the failure of the MPC confinement boundary welds. Therefore, leakage tests are not

required as part of the storage maintenance program.

### 9.2.3 Subsystem Maintenance

The HI-STORM 100 System does not include any subsystems which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, helium backfill, and leakage testing systems. Rigging, remote welders, cranes, and lifting beams shall also be inspected prior to each loading campaign to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations with the HI-STORM 100 require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also FSAR Section 9.2.6.

### 9.2.4 Pressure Relief Valves

The pressure relief valves used on the water jackets for the HI-TRAC transfer cask shall be calibrated on an annual basis (or prior to the next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year) to ensure pressure relief settings are 60 +2/-0 psig and 65 +2/-0 psig, or replaced with factory-set relief valves.

### 9.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM overpack, HI-TRAC, and MPC degrade negligibly over time or as a result of usage. To ensure continuing compliance of the HI-STORM 100 System to the design basis dose rate values, a shielding effectiveness test shall be performed every five years after placement into service.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.106(b) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC water jacket shall be verified prior to each loading campaign in accordance with the licensee's approved operations procedures.

The ~~Boral~~ *neutron absorber* panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. The use of Boral in similar nuclear applications is discussed in Chapter 1, and the long-term performance in a dry, inert gas atmosphere is evaluated in Chapter 3. *A similar discussion is provided for METAMIC<sup>®</sup> neutron absorber material.* Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM 100 System.

## 9.2.6 Thermal

In order to assure that the HI-STORM 100 System continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with ~~the Technical Specifications and~~ written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 9.2.1

## HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

<b>Task</b>	<b>Frequency</b>
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM 100 Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading, and every five years thereafter under the Maintenance Program
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC lifting trunnion and pocket trunnion recess visual inspection	Prior to each handling campaign
Load Testing of HI-TRAC Lifting Trunnions	Annually <sup>†</sup>
HI-TRAC pressure relief valve calibration	Annually <sup>†</sup>
HI-TRAC internal and external visual inspection for compliance to design drawings	Annually <sup>†</sup>
HI-TRAC water jacket water level visual examination	Prior to each handling campaign
HI-TRAC and Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

<sup>†</sup> Or prior to next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year.

### 9.3 REGULATORY COMPLIANCE

Chapter 9 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STORM 100 System in accordance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STORM 100 System will provide assurance that the SSCs important to safety will perform their design function. The performance of the maintenance program by the licensee for each loaded HI-STORM 100 System will provide assurance for the continued safe long-term storage of the stored SNF.

The described acceptance criteria and maintenance programs can be summarized in the following evaluation statements:

1. Section 9.1 of this FSAR describes Holtec International's proposed program for pre-operational testing and initial operations of the HI-STORM 100 System. Section 9.2 describes the proposed HI-STORM 100 maintenance program.
2. Structures, systems, and components (SSCs) of the HI-STORM 100 System designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.2.6 and 8.1.6 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STORM 100 System and its ancillary equipment, respectively. Tables 2.2.6 and 2.2.7 present the applicable standards for their design, fabrication, and inspection of the HI-STORM 100 System components.
3. Holtec International will examine and test the HI-STORM 100 System to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 9.1 of this FSAR describes the MPC confinement boundary assembly, inspection, and testing.
4. Holtec International will mark the cask with a data plate indicating its model number, unique identification number, and empty weight.
5. It can be concluded that the acceptance tests and maintenance program for the HI-STORM 100 System are in compliance with 10CFR72 [9.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STORM 100 System will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

## 9.4 REFERENCES

- [9.0.1] U.S. Code of Federal Regulations, Title 10, "Energy", Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste".
- [9.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.
- [9.1.1] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 1995 Edition, including Addenda through 1997.
- [9.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.
- [9.1.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.
- [9.1.4] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.
- [9.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5, January 1997.
- [9.1.6] Holtec International Position Paper DS-213, "Acceptable Flaw Size in MPC Lid-to-Shell Welds", Revision 2.

## CHAPTER 10: RADIATION PROTECTION<sup>†</sup>

This chapter discusses the design considerations and operational features that are incorporated in the HI-STORM 100 Storage System design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, transfer, and on-site dry storage. Occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections are provided. An off-site dose assessment for a typical ISFSI is also discussed. Since the determination of off-site doses is necessarily site-specific, similar dose assessments are to be prepared by the licensee, as part of implementing the HI-STORM 100 Storage System in accordance with 10CFR72.212 [10.0.1]. The information provided in this chapter meets all requirements of NUREG-1536.

### 10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

#### 10.1.1 Policy Considerations

The HI-STORM 100 has been designed in accordance with 10CFR72 [10.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [10.1.1] and the guidance provided in Regulatory Guides 8.8 [10.1.2] and 8.10 [10.1.3]. Licensees using the HI-STORM 100 System will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [10.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM 100 System, and be familiarized with the expected dose rates around the MPC, HI-STORM and HI-TRAC during all phases of loading, storage, and unloading operations. Chapter 12 provides dose rate limits at the HI-TRAC and HI-STORM surfaces to ensure that the HI-STORM 100 System is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings should be held with workers and radiological protection personnel prior to work on or around the system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities, will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [10.1.1] standards for radiation protection are met in accordance with the site's written commitments. Users can further reduce dose rates around the HI-STORM 100 System by preferentially loading longer-cooled and lower-burnup spent fuel assemblies in the periphery fuel storage cells of the MPC, and loading assemblies with shorter cooling times and higher burnups in the inner MPC fuel storage cell locations. Users can also further reduce the dose rates around the HI-TRAC by the use of temporary shielding. In some cases, users may opt to upgrade their existing crane to take advantage of the increased shielding capabilities of the 125-Ton HI-TRAC transfer cask (versus the 100-Ton HI-TRAC transfer cask). This decision should be based on a cost-benefit analysis. Temporary shielding and use of special tools to reduce dose is discussed in Section 10.1.4.

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



### 10.1.2 Design Considerations

Consistent with the design criteria defined in Section 2.3.5, the radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM 100 Storage System are as follows:

1. 10CFR72.104 [10.0.1] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements.
2. 10CFR72.106 [10.0.1] requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.
3. 10CFR20 [10.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.
4. Regulatory Position 2 of Regulatory Guide 8.8 [10.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM 100 Storage System as described below:
  - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [10.0.1]. Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a loaded HI-STORM 100 Storage cask is placed in an ISFSI. Due to the nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.
  - Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 or later in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM 100 System design include:
    - system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;

- system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;
  - system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;
  - system designs that minimize planned maintenance requirements;
  - system designs that minimize decontamination requirements at ISFSI decommissioning;
  - system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;
  - thick walled overpack that provides gamma and neutron shielding;
  - thick MPC lid which provides effective shielding for operators during MPC loading and unloading operations;
  - multiple welded barriers to confine radionuclides;
  - smooth surfaces to reduce decontamination time;
  - minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
  - capability of maintaining water in the MPC during welding to reduce dose rates;
  - capability of maintaining water in the transfer cask annulus space and water jacket to reduce dose rates during closure operations;
  - MPC penetrations located and configured to reduce streaming paths;
  - HI-STORM and HI-TRAC designed to reduce streaming paths;
  - MPC vent and drain ports with resealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;
  - use of a separate pool lid, annulus seal, and Annulus Overpressure System to prevent contamination of the MPC shell outer surfaces during in-pool activities;
  - temporary and auxiliary shielding to reduce dose rates around the HI-TRAC; and
  - low-maintenance design to reduce doses during storage operation.
- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.

- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM 100 Storage System is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this FSAR, and no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean water in the HI-TRAC transfer cask-MPC annulus and by using an inflatable annulus seal.
- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM 100 Storage System ISFSI since there are no radioactive systems at an ISFSI that could transport crud.
- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the HI-TRAC transfer cask is designed for ease of decontamination. In addition, an inflatable annulus seal is used to prevent fuel pool water from contacting and contaminating the exterior surface of the MPC.
- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.
- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins.
- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC shell, the primary confinement boundary. This material is resistant to the damaging effects of radiation and is well proven in the SNF cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

### 10.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STORM 100 System include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated welding system, lift yoke, transfer slide and moisture removal systems to reduce time operators spend in the vicinity of the loaded MPC;
- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;

- low fuel assembly lift-over height of the HI-TRAC maximizes water coverage over assemblies during fuel assembly loading;
- a water-filled neutron shield jacket allows filling after removal of the HI-TRAC from the spent fuel pool. This maximizes the shielding on the HI-TRAC without exceeding the crane capacity;
- descriptive operating procedures that provide guidance to reduce equipment contamination, obtain survey information, minimize dose and alert workers to possible changing radiological conditions;
- preparation and inspection of the HI-STORM and HI-TRAC in low-dose areas;
- MPC lid fit tests and inspections prior to actual loading to ensure smooth operation during loading;
- gas sampling of the MPC and HI-STAR 100 annulus (receiving from transport) to assess the condition of the cladding and MPC confinement boundary;
- fuel cool-down operations developed for fuel unloading operations which minimize thermal shock to the fuel and therefore reduce the potential for fuel cladding rupture;
- HI-STORM vent temperature elements (See Chapter 12) allow remote monitoring of the vent operability surveillance;
- wetting of component surfaces prior to placement in the spent fuel pool to reduce the need for decontamination;
- decontamination practices which consider the effects of weeping during HI-TRAC transfer cask heat up and surveying of HI-TRAC prior to removal from the fuel handling building;
- a sequence of operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations.

#### 10.1.4 Auxiliary/Temporary Shielding

To minimize occupational dose during loading and unloading operations, a specially-designed set of auxiliary shielding is available. The HI-STORM 100 auxiliary shielding consists of the Automated Welding System Baseplate, the HI-TRAC Temporary Shield Ring, the annulus shield, HI-STORM vent shield insert, the HI-TRAC transfer step, and the shield panel trim plates. Additional supplemental shielding such as lead blankets and bricks or other such shielding may also be used to help reduce dose rates. Each auxiliary shield is described in Table 10.1.1, shown on Figure 10.1.1 and the procedures for utilization are provided in Chapter 8. Other embodiments of the temporary shielding may also be used. Table 10.1.2 provides the minimum requirements for use of the temporary shielding indicating optional and required shielding. Users shall evaluate the need for auxiliary and temporary shielding and use of special

tooling to reduce the overall exposure based on an ALARA review of cask loading operations and the MPC contents.

Table 10.1.1  
HI-STORM 100 AUXILIARY AND TEMPORARY SHIELDS

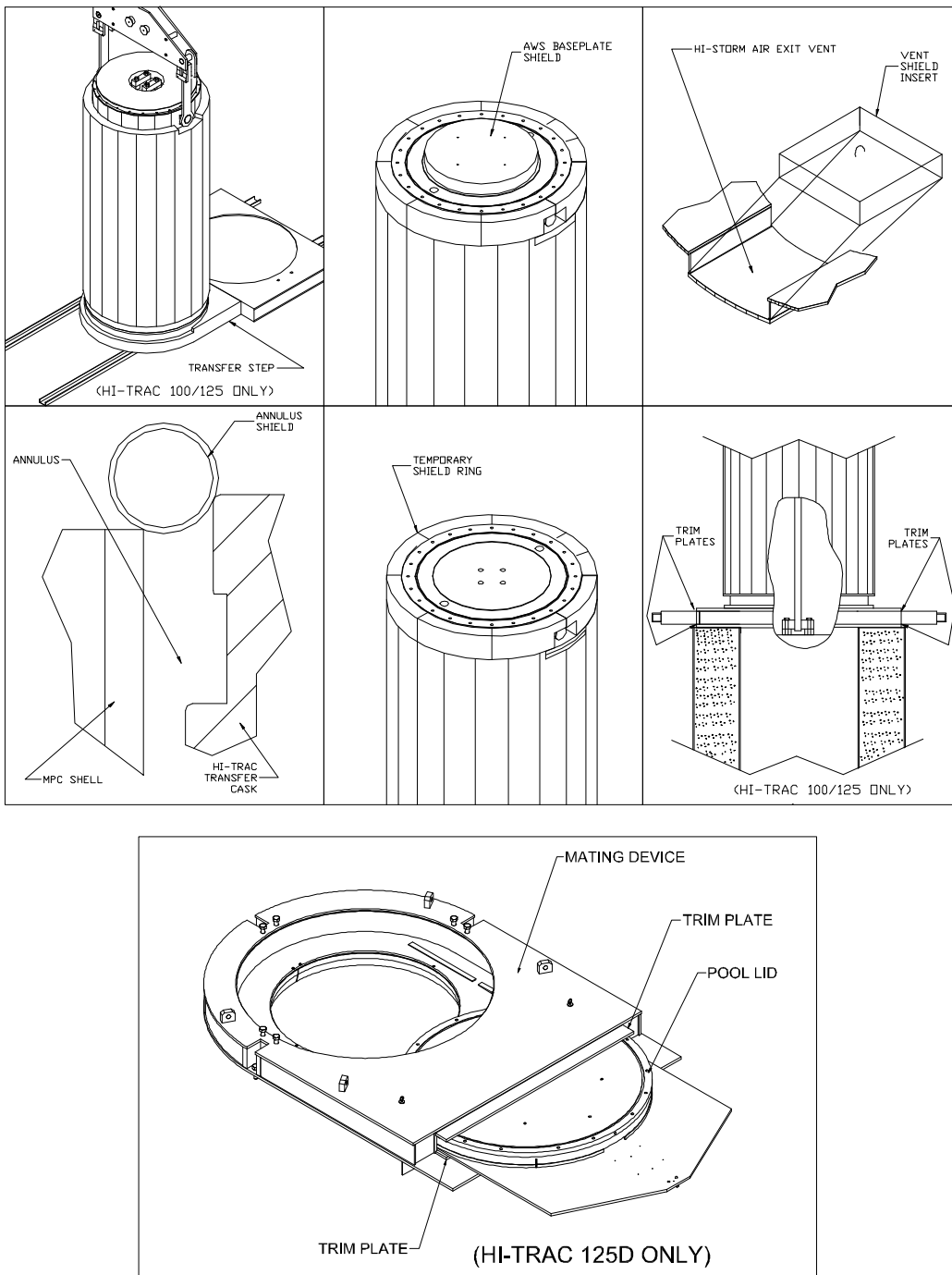
Temporary Shield	Description	Utilization
Automated Welding System Baseplate	Thick gamma and neutron shield circular plate that sits on the MPC lid. Plate is set directly on the MPC. Threaded lift holes are provided to assist in rigging.	Used during MPC closure and unloading operations in the cask preparation area to reduce the dose rates around the MPC lid. The design of the closure ring allows the baseplate shield to remain in place during the entire closure operation.
HI-TRAC Temporary Shield Ring	A water-filled tank that is placed atop of the HI-STAR or HI-TRAC neutron shield.	Used during MPC and HI-TRAC closure operations and MPC transfers into HI-STAR to reduce dose rates to the operators around the top flange of the HI-TRAC.
Annulus Shield	A shield that is seated between the MPC shell and the HI-TRAC.	Used during MPC closure operations to reduce streaming from the annulus.
HI-TRAC Transfer Step	A stepped block used to position the pool lid and transfer lid at the same elevation. The transfer step creates a tight seam between the two lids to eliminate streaming during bottom lid replacement.	Used during HI-TRAC bottom lid replacement.
Shield Panel Trim Plates	Four steel plates approximately 0.25 inch by 3 inch by 80 inch that are placed at the ends of the transfer lid top and bottom plate and secured by clamps or other method deemed suitable by the user.	Used during MPC transfer to and from HI-TRAC to shield the small gap above and below the sliding doors on the transfer lid.
HI-STORM Vent Shield Inserts	Devices shaped to fit into the HI-STORM exit vents.	Used during MPC transfer to and from HI-STORM to eliminate the streaming path from the exit vents during MPC transfer operations.

Table 10.1.2  
 HI-STORM 100 AUXILIARY AND TEMPORARY SHIELD REQUIREMENTS

<b>Auxiliary Shielding</b>	<b>Required for the 100-Ton HI-TRAC</b>	<b>Required for the 125-ton HI-TRAC and HI-TRAC 125D</b>
<b>Temporary Shield Ring</b>	Note 1	Note 1
<b>Automated Welding System Baseplate Shield</b>	No	No
<b>Annulus Shield</b>	Note 1	Note 1
<b>Vent Duct Shield Inserts</b>	Note 2	Note 2
<b>Transfer Step</b>	Yes	Yes (Note 3)
<b>Trim Plates</b>	No	No

Notes:

1. Users shall determine the need for this temporary shielding based on the specific operations and the MPC contents.
2. Not required for the HI-STORM 100S Overpack.
3. Not used with the HI-TRAC 125D.



**Figure 10.1.1; HI-STORM 100 System Auxiliary/Temporary Shielding**



## 10.2 RADIATION PROTECTION DESIGN FEATURES

The development of the HI-STORM 100 System has focused on design provisions to address the considerations summarized in Sections 10.1.2 and 10.1.3. The intent has been to improve on past concrete-based dry storage system designs by developing HI-STORM 100 as a hybrid of current metal and concrete storage system technologies. The design is, therefore, an evolution in storage systems, which incorporates preferred features from concrete storage, canister-based systems while retaining several of the advantages of metal casks as well. This approach results in a reduction in the need for maintenance, in overall radiation levels, and in the time spent on maintenance, when compared with current concrete-based dry storage systems. The following specific design features ensure a high degree of confinement integrity and radiation protection:

- HI-STORM 100 has been designed to meet storage condition dose rates required by 10CFR72 [10.0.1] for ~~five~~three-year cooled fuel;
- HI-STORM 100 has been designed to accommodate a maximum number of PWR or BWR fuel assemblies to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site;
- HI-STORM 100 overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance to corrosion and other forms of degradation (e.g., erosion);
- HI-STORM 100 has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the confinement boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or credible accident conditions;
- HI-TRAC transfer cask has a transfer step and other auxiliary shielding devices which eliminates streaming paths and simplify operations;
- The pool lid maximizes available fuel assembly water coverage in the spent fuel pool.
- The transfer lid is designed for quick alignment with HI-STORM; and
- HI-STORM 100 has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel.

### 10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM system. This section uses the shielding analysis provided in Chapter 5 and the operations procedures provided in Chapter 8 to develop a dose assessment. The dose assessment is provided in Tables 10.3.1, 10.3.2, and 10.3.3.

The dose rates from the HI-STORM 100 overpack, MPC lid, HI-TRAC transfer cask, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the various loading and unloading operations. The dose rates are also calculated for the various conditions of the cask that may affect the dose rates to the operators (e.g., MPC water level, HI-TRAC annulus water level, neutron shield water level, presence of temporary shielding). The dose rates around the 100-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of ~~42,500~~46,000 MWD/MTU and cooling of 53 years including BPRAs. The dose rates around the 125-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of ~~57,500~~75,000 MWD/MTU and cooling of ~~12-5~~ years including BPRAs. The dose rates around the HI-STORM 100 overpack are based on 24 PWR fuel assemblies with a burnup of ~~5247,500~~ MWD/MTU and cooling of ~~5-3~~ years. The selection of these fuel assembly types in all fuel cell locations bound all possible PWR and BWR loading scenarios for the HI-STORM System from a dose-rate perspective. No assessment is made with respect to background radiation since background radiation can vary significantly by site. In addition, exposures are based on work being performed with the temporary shielding described in Table 10.1.2.

The choice of burnup and cooling times used in this chapter is extremely conservative. The bounding burnup and cooling time that resulted in the highest dose rates around the 100-ton and 125-ton HI-TRACs were used in conjunction with the very conservative burnup and cooling time for the HI-STORM 100 overpack (as discussed in Section 5.1). In addition, including the source term from BPRAs increases the level of conservatism. The maximum dose rate due to BPRAs was used in this analysis. As stated in Chapter 5, using the maximum source for the BPRAs in conjunction with the bounding burnup and cooling time for fuel assemblies is very conservative as it is not expected that burnup and cooling times of the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values. This combined with the already conservative dose rates for the HI-TRACs and HI-STORMs results in an upper bound estimate of the occupational exposure. Users' radiation protection programs will assure appropriate temporary shielding is used based on actual fuel to be loaded and resulting dose rates in the field.

For each step in Tables 10.3.1 through 10.3.3, the operator work location is identified. These correspond to the locations identified in Figure 10.3.1. The relative locations refer to both the HI-STORM 100 Overpack and the HI-STORM 100s Overpack. The dose rate location points around the transfer cask and overpack were selected to model actual worker locations and cask conditions during the operation. Cask operators typically work at an arms-reach distance from the cask. To account for this, an 18-inch distance was used to estimate the dose rate for the

worker. This assessment addresses only the operators that perform work on or immediately adjacent to the cask.

Justification for the duration of operations along with the corresponding procedure steps from Chapter 8 are also provided in the tables. The assumptions used in developing time durations are based on mockups of the MPC, review of design drawings, walk-downs using other equipment to represent the HI-TRAC transfer cask and HI-STORM 100 overpack the HI-STAR 100 overpack and MPC-68 prototype, consultation with UST&D (weld examination) and consultation with cask operations personnel from Calvert Cliffs Nuclear Power Plant (for items such as lid installation and decontamination). In addition, for the shielding calculations, only the Temporary Shield Ring was assumed to be in place for applicable portions of the operations.

Tables 10.3.1a, 10.3.1b, and 10.3.1c provide a summary of the dose assessment for a HI-STORM 100 System loading operation using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and the 125-ton HI-TRAC 125D respectively. Tables 10.3.2a, 2b, and 2c provide a summary of the dose assessment for HI-STORM 100 System unloading operations using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and 125-ton HI-TRAC 125D respectively. Tables 10.3.3a, 3b, and 3c provide a summary of the dose assessment for transferring the MPC to a HI-STAR 100 overpack as described in Section 8.5 of the operating procedures using the 125-ton HI-TRAC and the 100-ton HI-TRAC transfer cask, respectively.

#### 10.3.1 Estimated Exposures for Loading and Unloading Operations

The assumptions used to estimate personnel exposures are conservative by design. The main factors attributed to actual personnel exposures are the age and burnup of the spent fuel assemblies and good ALARA practices. To estimate the dose received by a single worker, it should be understood that a canister-based system requires a diverse range of disciplines to perform all the necessary functions. The high visibility and often critical path nature of fuel movement activities have prompted utilities to load canister systems in a round-the-clock mode in most cases. This results in the exposure being spread out over several shifts of operators and technicians with no single shift receiving a majority of the exposure.

The total person-rem exposure from operation of the HI-STORM 100 System is proportional to the number of systems loaded. A typical utility will load approximately four MPCs per reactor cycle to maintain the current available spent fuel pool capacity. Utilities requiring dry storage of spent fuel assemblies typically have a large inventory of spent fuel assemblies that date back to the reactor's first cycle. The older fuel assemblies will have a significantly lower dose rate than the design basis fuel assemblies due to the extended cooling time (i.e., much greater than the values used to compute the dose rates). Users shall assess the cask loading for their particular fuel types (burnup, cooling time) to satisfy the requirements of 10CFR20 [10.1.1].

For licensees using the 100-Ton HI-TRAC transfer cask, design basis dose rates will be higher (than a corresponding 125-Ton HI-TRAC) due to the decreased mass of shielding. Due to the higher expected dose rates from the 100-Ton HI-TRAC, users may need to use the auxiliary shielding (See Table 10.1.2), and should consider preferential loading, and increased precautions

(e.g., additional temporary or auxiliary shielding, remotely operated equipment, additional contamination prevention measures). Actual use of optional dose reduction measures must be decided by each user based on the fuel to be loaded.

### 10.3.2 Estimated Exposures for Surveillance and Maintenance

Table 10.3.4 provides ~~the maximum~~ *an estimate of the* occupational exposure required for security surveillance and maintenance of an ISFSI. ~~Although the HI-STORM 100 System requires only minimal maintenance during storage, maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. The estimated dose rates described below are based on a sample array of HI-STORM 100 overpacks fully loaded with design basis fuel assemblies, placed at their minimum required pitch, in a 2 x 6 HI-STORM array. The maintenance worker is assumed to be at a distance of 5 meters from the center of the long edge of the array. The security worker is assumed to be at a distance of 15 meters from the center of the long edge of the array. Users may opt to utilize electronic temperature monitoring of the HI-STORM modules or remote viewing methods instead of performing direct visual observation of the modules. Since security surveillances can be performed from outside the ISFSI and since the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area, a dose rate of 3 mrem/hour is estimated. Although the HI-STORM 100 System requires only minimal maintenance during storage (e.g. touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Since most of the maintenance is expected to occur outside the actual cask array, For maintenance of the casks and the ISFSI, a dose rate of 10 mrem/hour is estimated~~

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (~~57,500~~75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
<del>CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM</del>	4	8A	4	37.9	2.5	2.5	SIMPLE ATTACHMENT NO TOOLS
<del>DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM</del>	4	8A	4	37.9	2.5	2.5	SIMPLE ATTACHMENT NO TOOLS
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACHM MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING
FLUSH CAVITY WITH HELIUM AND INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
<del>INSTALL MSLD OVER VENT PORT COVER PLATE</del>	<del>2</del>	<del>8A</del>	<del>1</del>	<del>37.9</del>	<del>1.3</del>	<del>1.3</del>	<del>INSTALLED BY HAND NO TOOLS</del>
<del>INSTALL MSLD OVER DRAIN PORT COVER PLATE</del>	<del>2</del>	<del>8A</del>	<del>1</del>	<del>37.9</del>	<del>1.3</del>	<del>1.3</del>	<del>INSTALLED BY HAND NO TOOLS</del>
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	354.2	106.3	106.3	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
<b>Section 8.1.7</b>							



**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	43.8	11.7	23.3	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	7.3	1.2	2.4	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN

**Table 10.3.1a**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500 75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>						<b>796.0 1,797.2804.8 PERSON-MREM</b>	

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,50046,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	3	3.4	6.8	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	3	2.3	4.5	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	3	1.0	2.0	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	76.3	7.6	15.3	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	663.4	94.0	94.0	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	432.5	72.1	72.1	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	919.1	76.6	76.6	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	241.8	2.0	2.0	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	76.3	12.7	12.7	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	76.3	2.5	2.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	919.1	30.6	30.6	SINGLE PLUG, NO SPECIAL TOOLS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	62.5	6.3	12.5	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	62.5	3.1	3.1	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	919.1	1577.8	1577.8	490 SQ-FT@5 SQ-FT/PERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	31.3	8.3	16.7	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	31.3	13.0	13.0	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	241.8	2.0	2.0	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	31.3	1.0	1.0	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	31.3	1.0	1.0	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	31.3	2.6	2.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	31.3	3.1	3.1	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	31.3	1.6	1.6	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	31.3	0.5	0.5	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	31.3	1.0	1.0	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	31.3	10.4	31.3	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	31.3	6.3	12.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	31.3	10.4	20.9	ALIGN AND REMOVE 4 SHACKLES

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	31.3	4.2	8.3	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	31.3	120.0	120.0	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	31.3	0.5	0.5	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	31.3	5.2	5.2	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	31.3	1.0	1.0	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	31.3	23.5	23.5	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	31.3	0.5	0.5	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	31.3	0.5	0.5	1" THREADED FITTING NO TOOLS
CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM	4	8A	1	60.0	4.0	4.0	SIMPLE ATTACHMENT NO TOOLS
DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM	4	8A	1	60.0	4.0	4.0	SIMPLE ATTACHMENT NO TOOLS
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	60.0	2.0	2.0	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM () TO VENT AND DRAIN PORT RVOAs	2	8A	1	60.0	2.0	2.0	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	60.0	2.0	2.0	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	60.0	1.5	1.5	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	60.0	1.5	1.5	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	60.0	2.0	2.0	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	60.0	1.0	1.0	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE VENT AND DRAIN COVER PLATE WELD	100	8A	1	60.0	100.0	100.0	MEASURED DURING WELD MOCKUP TESTING
FLUSH CAVITY WITH HELIUM AND INSTALL SET SCREWS	2	8A	1	60.0	2.0	2.0	4 SET SCREWS @2/MINUTE
PLUG WELD OVER ET SCREWS	8	8A	1	60.0	8.0	8.0	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
<del>INSTALL MSLD OVER VENT PORT COVER PLATE</del>	2	8A	1	60.0	2.0	2.0	INSTALLED BY HAND NO TOOLS
<del>INSTALL MSLD OVER DRAIN PORT COVER PLATE</del>	2	8A	1	60.0	2.0	2.0	INSTALLED BY HAND NO TOOLS
INSTALL AND ALIGN CLOSURE RING	5	8A	1	60.0	5.0	5.0	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	60.0	185.0	185.0	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	60.0	12.0	12.0	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	60.0	1.0	1.0	SHIELD PLACED BY HAND

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH DRAIN LINE TO HI-TRAC POSITION HI-TRAC TOP LID	1	9D	1	1806.3	30.1	30.1	1" THREADED FITTING NO TOOLS
TORQUE TOP LID BOLTS	10	9B	2	60.0	10.0	20.0	VERTICAL FLANGED CONNECTION
INSTALL MPC LIFT CLEATS AND MPC SLINGS	12	9B	1	60.0	12.0	12.0	24 BOLTS AT 2/MIN (INSTALL AND TORQUE, 1 PASS)
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	25	9A	2	247.7	103.2	206.4	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING SEGMENTS	1	9B	1	60.0	1.0	1.0	8 PLUGS @ 8/MIN
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	1	247.7	16.5	16.5	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
POSITION HI-TRAC ABOVE TRANSFER STEP	4	9A	2	247.7	16.5	33.0	INSTALLED BY HAND NO TOOLS
REMOVE BOTTOM LID BOLTS	15	9C	1	740.6	185.2	185.2	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
INSTALL TRANSFER LID BOLTS	6	10A	1	1806.3	180.6	180.6	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
DISCONNECT MPC SLINGS	18	11B	1	1806.3	541.9	541.9	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
<b>Section 8.1.7</b>							
POSITION HI-TRAC ON TRANSPORT DEVICE	4	9A	2	247.7	16.5	33.0	INSTALLED BY HAND NO TOOLS
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	20	11A	2	740.6	246.9	493.7	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
MATE OVERPACKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13B	2	561.8	93.6	187.3	ALIGNMENT GUIDES USED
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING NO TOOLS
INSTALL TRIM PLATES	4	13B	2	561.8	37.5	74.9	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	4	13B	2	561.8	37.5	74.9	INSTALLED BY HAND
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING
	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING

**Table 10.3.1b**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	43.8	11.7	23.3	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	7.3	1.2	2.4	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>						<b>2755.05,210.822.8 PERSON-MREM</b>	



**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.4</b>							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1.0	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
<b>Section 8.1.5</b>							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PRPERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PERFORM NDE OF LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
<del>CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM</del>	4	8A	1	37.9	2.5	2.5	SIMPLE ATTACHMENT NO TOOLS
<del>DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM</del>	4	8A	1	37.9	2.5	2.5	SIMPLE ATTACHMENT NO TOOLS
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING
FLUSH CAVITY WITH HELIUM AND INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINUTES
<del>INSTALL MSLD OVER VENT PORT COVER PLATE</del>	2	8A	1	37.9	1.3	1.3	INSTALLED BY HAND NO TOOLS
<del>INSTALL MSLD OVER DRAIN PORT COVER PLATE</del>	2	8A	1	37.9	1.3	1.3	INSTALLED BY HAND NO TOOLS
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM A NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.1.6</b>							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE, 1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND, NO TOOLS
<b>Section 8.1.7</b>							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
ATTACH MPC LIFT SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE MATING DEVICE LOCKING PINS AND OPEN DRAWER	40	13B	2	118.5	79.0	158.0	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS, NO TORQUING

**Table 10.3.1c**  
**HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE MATING DEVICE	10	15A	1	45.5	7.6	7.6	3 BOLTS @ 2 MINUTES PER BOLT
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	43.8	11.7	23.3	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	7.3	1.2	2.4	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASUREMENTS@1/MIN
<b>TOTAL</b>							<b><del>779.3</del>1751.79.3 PERSON-MREM</b>

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500 75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	7.3	1.2	1.2	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	73.9	9.8	9.8	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	354.2	106.3	106.3	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	158.5	26.4	26.4	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER



**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)

**Table 10.3.2a**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (~~57,500~~75,000 MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
<b>TOTAL</b>						<b>380.0809.5 PERSON-MREM</b>	

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	7.3	1.2	1.2	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	73.9	9.8	9.8	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	561.8	93.6	187.3	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	561.8	37.5	74.9	INSTALLED BY HAND NO FASTENERS

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	247.7	41.3	41.3	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	561.8	37.5	74.9	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	740.6	246.9	493.7	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	247.7	16.5	33.0	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	1806.3	180.6	180.6	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	1806.3	541.9	541.9	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	247.7	41.3	41.3	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	740.6	185.2	185.2	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	60.0	6.0	6.0	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	31.3	1.0	1.0	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	1806.3	15.1	15.1	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	60.0	9.0	9.0	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	31.3	20.9	41.7	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	60.0	1.0	1.0	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	60.0	2.0	2.0	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	60.0	0.5	0.5	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	60.0	0.5	0.5	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	60.0	2.0	2.0	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	60.0	1.0	1.0	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	31.3	5.2	5.2	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	60.0	1.0	1.0	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	31.3	5.2	10.4	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	241.8	8.1	8.1	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	241.8	8.1	8.1	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	62.5	2.1	2.1	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	62.5	12.5	12.5	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	241.8	2.0	2.0	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	241.8	40.3	40.3	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	663.4	94.0	94.0	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	76.3	15.3	15.3	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	3	1.0	2.0	40 FEET @ 2 FT/MINUTE (CRANE SPEED)

**Table 10.3.2b**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (~~42,500~~46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LID	20	2	2	3	1.0	2.0	CONSULTATION WITH CALVERT CLIFFS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
<b>TOTAL</b>						<b>1387.72569.7 PERSON-MREM</b>	

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)</b>							
REMOVE INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	43.8	3.6	7.3	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	7.3	1.2	1.2	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	73.9	9.8	9.8	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
INSTALL MATING DEVICE WITH POOL LID	10	15A	1	45.5	7.6	7.6	3 BOLTS AT 2 MINUTES PER BOLT
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE MATING DEVICE DRAWER AND BOLT-UP POOL LID	36	13B	2	118.5	71.1	142.2	2 PINS@2MIN/PIN, 16 BOLTS @ 2MIN/BOLT
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
<b>Section 8.3.3</b>							
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER



**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @ 10 FT/MIN (CRANE SPEED)

**Table 10.3.2c**  
**HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (~~57,500~~75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS
<b>Section 8.3.4</b>							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
<b>TOTAL</b>						<b>329.1672.6 PERSON-MREM</b>	

**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (~~57,500~~75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	7.1	1.2	2.4	ALIGN AND POSITION REMOVE 4 SHACKLES

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	362.5	151.0	302.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	118.5	7.9	15.8	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN

**Table 10.3.3a**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500 75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	43.8	11.7	11.7	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	7.3	1.2	1.2	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	43.8	3.6	3.6	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASMT@1/MIN
<b>TOTAL</b>						<b>465.4722.6 PERSON-MREM</b>	

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	7.1	1.2	2.4	ALIGN AND POSITION REMOVE 4 SHACKLES

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,50046,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	362.5	151.0	302.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	561.8	93.6	187.3	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	561.8	37.5	74.9	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	561.8	37.5	74.9	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	561.8	37.5	74.9	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	247.7	41.3	82.6	2 SLINGS@5/MIN
REMOVE TRIM PLATES	4	13B	2	561.8	37.5	74.9	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	45.5	3.0	3.0	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN

**Table 10.3.3b**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 100-TON HI-TRAC TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (42,500/46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	43.8	11.7	11.7	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	7.3	1.2	1.2	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	43.8	3.6	3.6	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASMT@1/MIN
<b>TOTAL</b>						<b>718.81136.5 PERSON-MREM</b>	



**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500/75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
<b>Section 8.5.2</b>							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES

<sup>†</sup> See notes at bottom of Table 10.3.4.

**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,50075,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL MATING DEVICE ON HI-STAR	20	19B	2	7.1	2.4	4.7	ALIGN AND BOLT INTO PLACE
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	362.5	151.0	302.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE LOCKING PINS AND OPEN DRAWER	4	20B	2	118.5	7.9	15.8	2 <a href="#">PINS@2/MIN</a>
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
<b>Section 8.5.3</b>							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
RAISE THE POOL LID AND BOLT INTO PLACE ON HI-TRAC	32	20B	2	118.5	63.2	126.4	2 MINS/BOLT, 16 BOLTS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 <a href="#">SLINGS@5/MIN</a>
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	362.5	12.1	12.1	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	45.5	1.5	1.5	4 SHACKLES@2/MIN
REMOVE THE MATING DEVICE	6	15A	1	45.5	4.5	4.5	3 BOLTS AT 2 MINUTES PER BOLTS

**Table 10.3.3c**  
**MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING**  
**THE 125-TON HI-TRAC 125D TRANSFER CASK**  
**ESTIMATED OPERATIONAL EXPOSURES<sup>†</sup> (57,500,75,000 MWD/MTU, 125-YEAR COOLED PWR FUEL)**

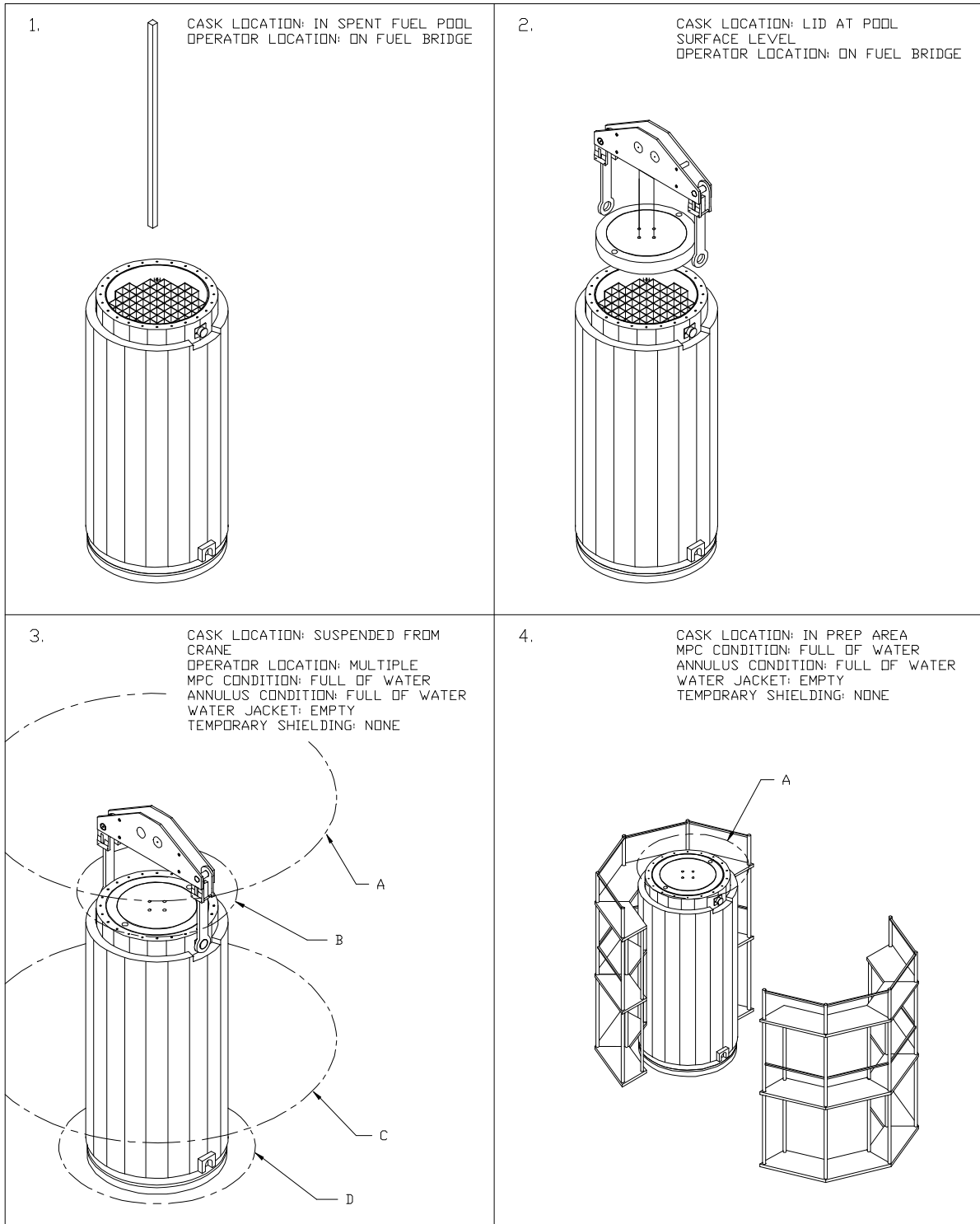
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	7.3	3.1	6.1	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	73.9	4.9	4.9	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	73.9	24.6	24.6	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	73.9	24.6	24.6	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	7.3	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	7.3	0.2	0.2	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	43.8	11.7	11.7	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	7.3	1.2	1.2	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	25.5	17.0	17.0	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS @1/MIN
REMOVE AIR PAD	5	16D	1	43.8	3.6	3.6	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	43.8	2.9	2.9	4 JACKS @1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	43.8	14.6	14.6	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	73.9	9.8	9.8	8 MEASMT@1/MIN
<b>TOTAL</b>							<b>-514.0852.9 PERSON-MREM</b>

Table 10.3.4  
ESTIMATED EXPOSURES FOR HI-STORM 100 SURVEILLANCE AND MAINTENANCE

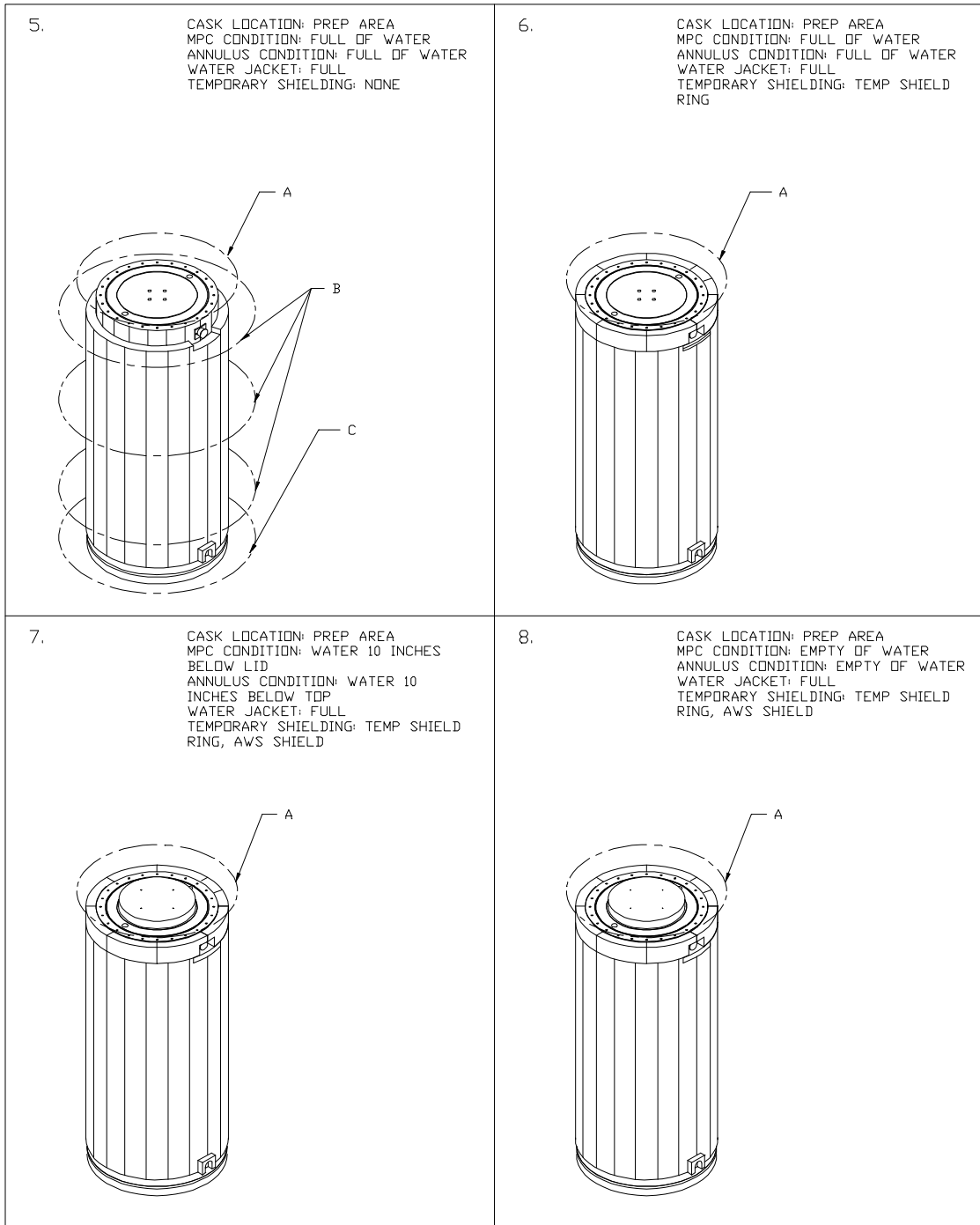
ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 10.3.1a, 10.3.1b, 10.3.1c, 10.3.2a, 10.3.2b, 10.3.2c, 10.3.3a, 10.3.3b, 10.3.3c and 10.3.4:

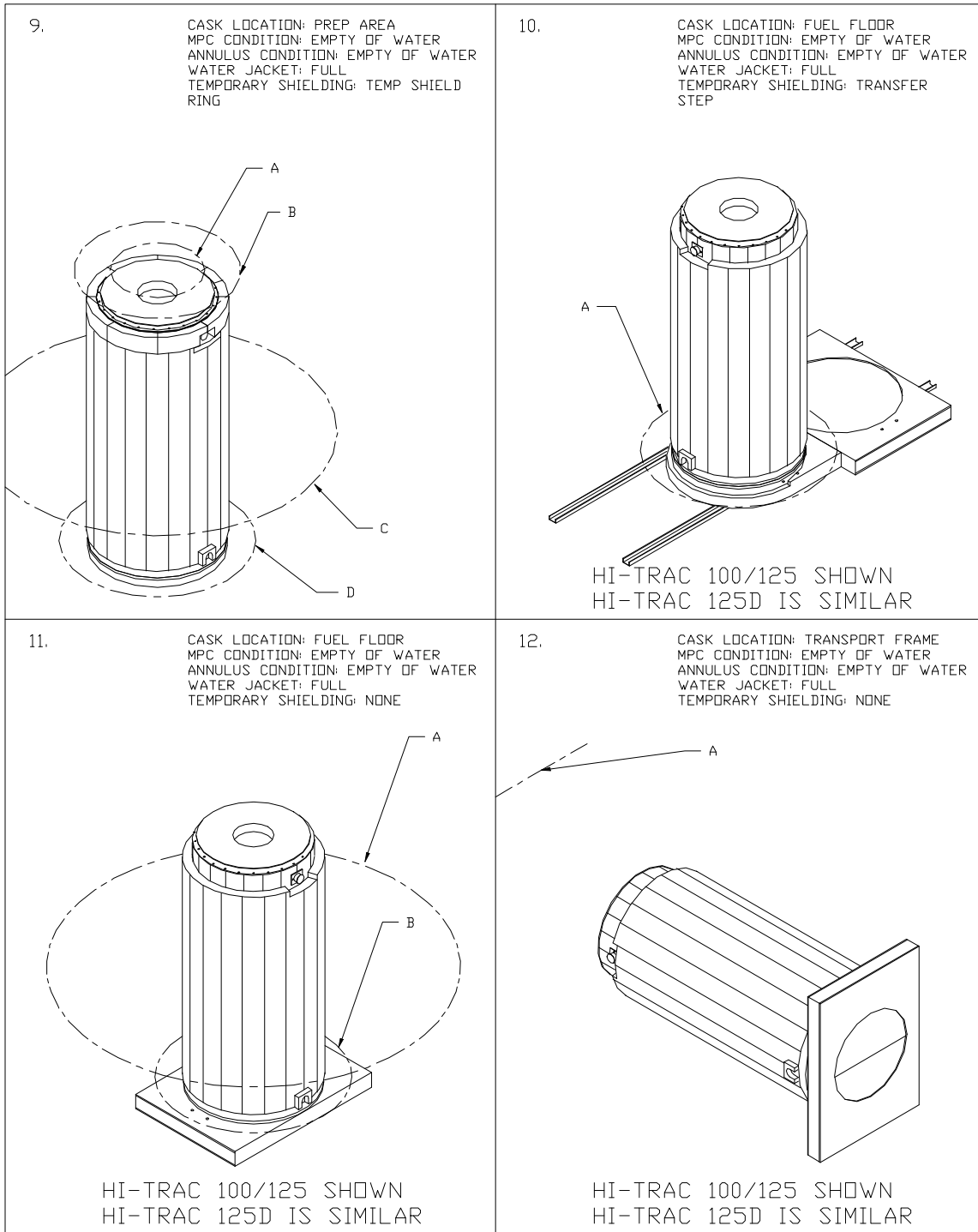
1. Refer to Chapter 8 for detailed description of activities.
2. Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.
3. HI-STAR 100 Operations assume that the cooling time is at least 10 years.



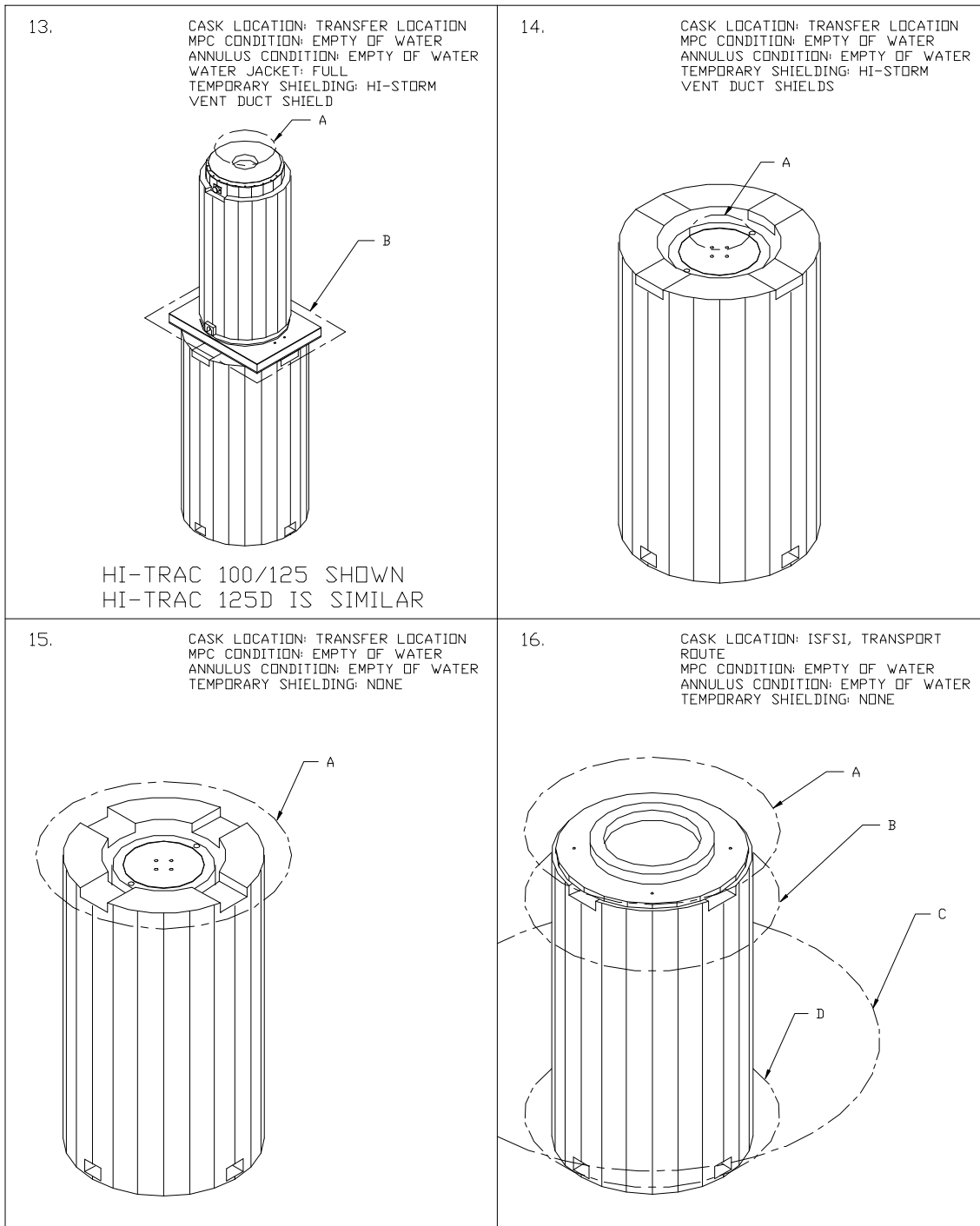
**Figure 10.3.1a; Operator Work Locations Used for Estimating Personnel Exposure**



**Figure 10.3.1b; Operator Work Locations Used for Estimating Personnel Exposure**

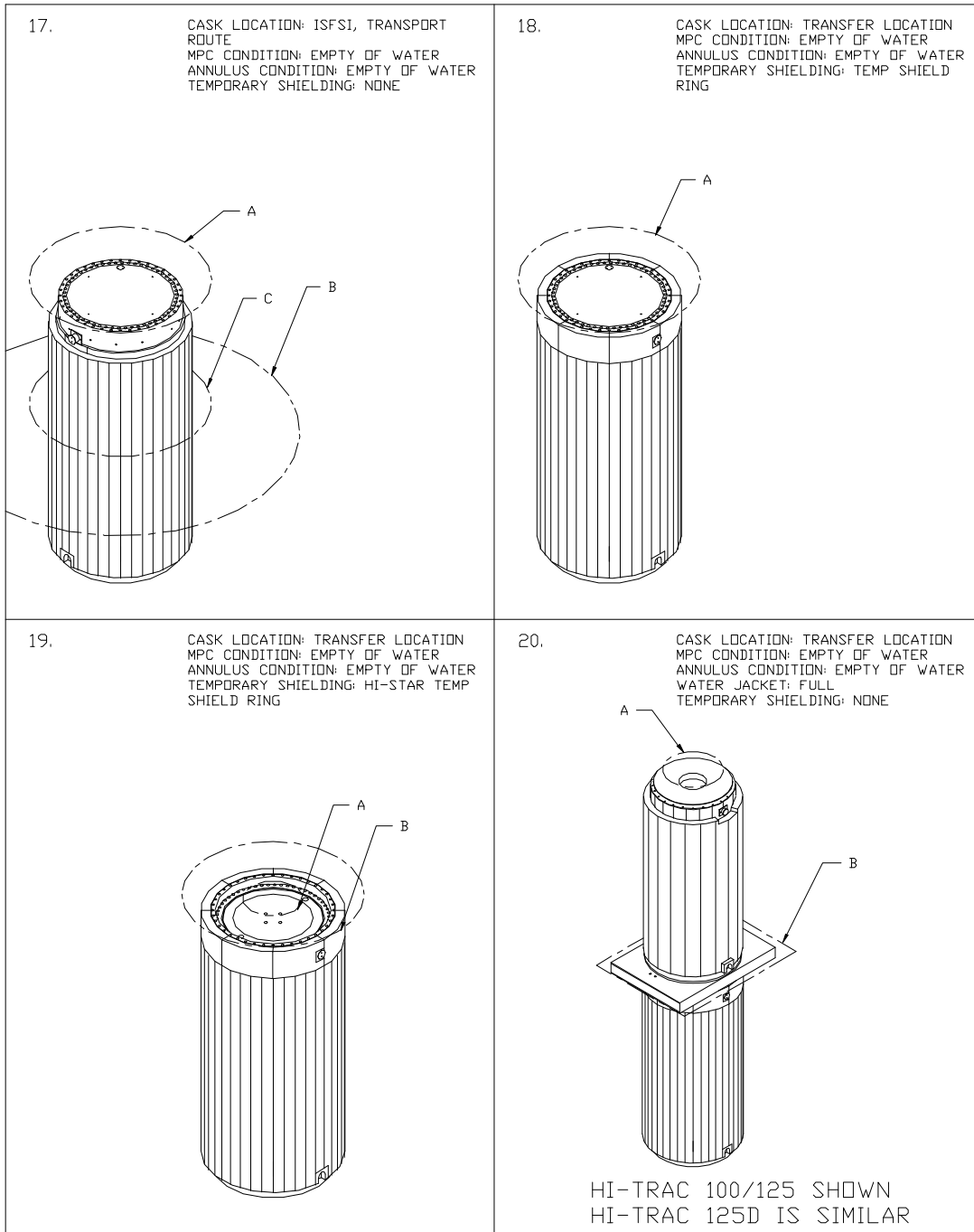


**Figure 10.3.1c; Operator Work Locations Used for Estimating Personnel Exposure**



**Figure 10.3.1d; Operator Work Locations Used for Estimating Personnel Exposure**





**Figure 10.3.1e; Operator Work Locations Used for Estimating Personnel Exposure**

## 10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT

### 10.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [10.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STORM 100 System with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10CFR72.212 [10.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.9 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [10.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. As a summary of Chapter 5, Table 10.4.1 presents the annual dose results for a single overpack at 100 and ~~200–250~~ meters and a 2x5 array of HI-STORM 100 systems at ~~350–450~~ meters. These annual doses are based on a full array of design basis fuel with a burnup of ~~5247,500~~ MWD/MTU and ~~53~~-year cooling. This burnup and cooling time combination conservatively bounds the allowable burnup and cooling times listed in ~~the Technical Specifications Appendix B to the CoC Section 2.1.9.~~ In addition, 100% occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, the casks were positioned on an infinite slab of soil to account for earth-shine effects. These results indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of ~~200–250~~ meters for a single cask and at ~~350–450~~ meters for a 2x5 array of HI-STORM 100 Systems containing design basis fuel. These results are presented only as an illustration to demonstrate that the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1]. Neither the distances nor the array configurations become part of the Technical Specifications. Rather, users are required to perform a site specific analyses to demonstrate compliance with 10CFR72.104[10.0.1] contributors and 10CFR20[10.1.1].

An additional contributor to the controlled area boundary dose is the loaded HI-TRAC transfer cask, if the HI-TRAC is to be used at the ISFSI outside of the fuel building. Table 10.4.2 provides dose rates at 100, 200, and 300 meters for a 100-ton HI-TRAC transfer cask loaded with design basis fuel. The 100-ton HI-TRAC dose rates bound the 125-ton HI-TRAC by large margins. Based on the short duration that the loaded HI-

TRAC is used outside at the ISFSI, the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1] when worst-case design basis fuel is loaded in all fuel cell locations. However, users are required to perform a site specific analysis to demonstrate compliance with 10CFR72.104[10.0.1] and 10CFR20[10.1.1] taking into account the actual site boundary distance and fuel characteristics.

*Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. A minor contributor to the minimum controlled area boundary is the normal storage condition leakage from the welded MPC. Although leakage is not expected, Section 7.2 provides an analysis for the annual dose equivalent based on a continuous leak from the MPC. The annual dose equivalent to an individual at the minimum controlled area boundary based on the assumed leakage rate and continuous occupancy is presented in Table 7.3.8. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.*

#### 10.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Section 11.1, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and off-normal handling of HI-TRAC) do not result in the degradation of the HI-STORM 100 System shielding effectiveness. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that of normal conditions.

~~However, the annual dose at the controlled area boundary as a result of an assumed effluent release under off normal conditions is different than that under normal conditions. Under off normal conditions, 10% of the fuel rods are assumed to have been breached, in lieu of 1% of the fuel rods for normal conditions. The resulting annual dose equivalent to an individual at the minimum controlled area boundary, based on the assumed leakage rate and continuous occupancy, is presented in Table 7.3.8. The analysis to determine the off normal dose at the controlled area boundary is described in Section 7.2.~~

#### 10.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies that the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 10.1.2). In

addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

~~Subsection 7.3 and Table 7.3.8 demonstrates that the resultant doses for a non-mechanistic postulated breach of the MPC confinement boundary at the regulatory minimum site boundary distance of 100 meters is presented in Table 7.3.8 within the regulatory limits specified in 10CFR72.106 [10.0.1].~~

Chapter 11 presents the results of the evaluations performed to demonstrate that the HI-STORM 100 System can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [10.0.1]. The accident events addressed in Chapter 11 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, confinement boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, partial blockage of MPC basket air inlets, and 100% blockage of air inlets.

The worst-case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-STORM overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded. However, with over 25 inches of concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-TRAC transfer cask assumes that as a result of a fire, tornado missile, or handling accident, all of the water in the water jacket is lost. The shielding analysis of the 100-ton HI-TRAC transfer cask with complete loss of the water from the water jacket is discussed in Section 5.1.2. These results bound those for the 125-Ton HI-TRAC transfer cask by a large margin. The results in that section show that the resultant dose rate at the 100-meter controlled area boundary would be approximately ~~1.474.07~~ mrem/hour for the loaded HI-TRAC transfer cask during the accident condition. At the calculated dose rate, it would take approximately ~~141.51~~ days for the dose at the controlled area boundary to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 11. Therefore, the dose requirement of 10CFR72.106 [10.0.1] is satisfied. Once again, this dose is calculated assuming design basis fuel in all fuel cell locations. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 10.4.1

ANNUAL DOSE FOR ARRAYS OF HI-STORM 100 OVERPACKS  
 WITH DESIGN BASIS ZIRCALOY CLAD FUEL  
 5247,500 MWD/MTU AND 53-YEAR COOLING

Array Configuration	1 Cask	1 Cask	2x5 Array
Annual Dose (mrem/year) <sup>†</sup>	<del>130.02</del> 307.9	<del>20.19</del> 24.10	<del>18.64</del> 16.29
Distance to Controlled Area Boundary (meters) <sup>††, †††</sup>	100	<del>200</del> 250	<del>350</del> 450

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† 100% occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 53-year cooling as specified in ~~the Technical Specifications~~ *Appendix B to the CoC Section 2.1.9* is lower than the burnup analyzed for the design basis fuel used in this table.

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Table 10.4.2  
DOSE RATE FOR THE 100-TON HI-TRAC TRANSFER CASK  
WITH DESIGN BASIS ZIRCALOY CLAD FUEL

<b>Fuel Burnup &amp; Cooling Time</b>	<b>100 Meters</b>	<b>200 Meters</b>	<b>300 Meters</b>
<b>42,500</b> <b>MWD/MTU &amp; 5-3</b> <b>Years</b>	0.430.98 mrem/hr	0.070.15 mrem/hr	0.020.04 mrem/hr
<b>52,500</b> <b>MWD/MTU &amp; 10</b> <b>5 Years</b>	0.270.80 mrem/hr	0.040.12 mrem/hr	0.010.03 mrem/hr

## 10.5 REFERENCES

- [10.0.1] *U.S. Code of Federal Regulations*, Title 10, "Energy" Part 72 "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."
- [10.1.1] *U.S. Code of Federal Regulations*, Title 10, "Energy" Part 20 "Standards for Protection Against Radiation."
- [10.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.
- [10.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May1997.

## CHAPTER 11<sup>†</sup>: ACCIDENT ANALYSIS

This chapter presents the evaluation of the HI-STORM 100 System for the effects of off-normal and postulated accident conditions. The design basis off-normal and postulated accident events, including those resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Sections 2.2.2 and 2.2.3. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, confinement, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM 100 System are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and evaluations described therein.

Chapter 11 is in full compliance with NUREG-1536; no exceptions are taken.

### 11.1 OFF-NORMAL CONDITIONS

~~During normal storage operations of the HI-STORM 100 System it is possible that an off-normal situation could occur.~~ Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are listed in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM 100:

- Off-Normal Pressures
- Off-Normal Environmental Temperatures
- Leakage of One MPC Seal Weld
- Partial Blockage of Air Inlets
- Off-Normal Handling of HI-TRAC Transfer Cask
- Failure of FHD System*
- SCS Power Failure*
- Off-Normal Loads<sup>‡</sup>*

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

<sup>‡</sup> *Off-normal load combinations are defined in Chapter 2, Table 2.2.14 and evaluated in Chapter 3, Section 3.4.*



For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events without affecting function, and are in compliance with the applicable acceptance criteria. The following sections present the evaluation of the HI-STORM 100 System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for off-normal conditions are defined in Table 2.2.14. The load combinations include both normal and off-normal loads. The off-normal load combination evaluations are discussed in Section 11.1.5.

#### 11.1.1 Off-Normal Pressures

The sole pressure boundary in the HI-STORM 100 System is the MPC ~~internal pressure boundary enclosure vessel~~. The off-normal pressure condition is specified in Section 2.2.2.1. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature obtained with maximum decay heat load design basis fuel. The maximum off-normal environmental temperature is 100°F with full solar insolation. The MPC internal pressure *is evaluated with* ~~is further increased by the conservative assumption that~~ 10% of the fuel rods ruptured and 100% of the *rods* fill gas; and 30% of the fission gases ~~are~~ released to the cavity.

##### 11.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is low. Nonetheless, the event is postulated and evaluated.

##### 11.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure and, therefore, no monitoring is required.

##### 11.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

Chapter 4 calculates the MPC internal pressure with an ambient temperature of 80°F, 10% fuel rods ruptured, full insolation, and maximum decay heat, and reports the maximum value of 75.0 psig in Table 4.4.14 at an average temperature of 513.6°K. Using this pressure, the off-normal temperature of 100°F (*bounding temperature rise* ~~ΔT~~ of 20°F or 11.1°K), and the ideal gas law, the off-normal resultant pressure ~~is~~ *(calculated below)* ~~to be~~ *is below the MPC off-normal design*

pressure (Table 2.2.1 in Chapter 2). ~~condition MPC internal design pressure.~~

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$
$$P_2 = \frac{P_1 T_2}{T_1}$$
$$P_2 = \frac{(75.0 \text{ psig} + 14.7) (513.6^\circ \text{ K} + 11.1^\circ \text{ K})}{513.6^\circ \text{ K}}$$
$$P_2 = 91.6 \text{ psia or } 76.9 \text{ psig}$$

*It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would decrease the temperature rise, the calculated pressure is higher than would actually occur. ~~The off-normal MPC internal design pressure of 100 psig (Table 2.2.1) has been established to bound the off-normal condition. Therefore, no additional analysis is required.~~*

### Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is equivalent to the evaluation at normal internal pressures, since the normal design pressure was set at a value which would encompass the off-normal pressure. Therefore, the resulting stresses from the off-normal condition are equivalent to that of the normal condition and are well within the short-term allowable values, as discussed in Section 3.4. *The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.*

### Thermal

The MPC internal pressure for off-normal conditions is calculated as presented above. As can be seen from the value above, the ~~100 psig~~ design basis internal pressure for off-normal conditions used in the structural evaluation (Table 2.2.1) bounds the calculated value above.

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100 System.

### 11.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There is no corrective action requirement for off-normal pressure.

### 11.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

### 11.1.2 Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (HI-STORM overpack) and 0 to 100°F (HI-TRAC transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM 100 System and must be evaluated against the allowable component design temperatures. ~~This~~*The off-normal event is of a short duration, therefore the resultant* temperatures are evaluated against the ~~accident~~ *off-normal* condition temperature limits as listed in Table 2.2.3.

#### 11.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

### 11.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM overpack and MPC. Chapter 2 provides operational limitations to the use of the HI-TRAC transfer cask at temperatures of  $\leq 32^{\circ}\text{F}$  and prohibits use of the HI-TRAC transfer cask below  $0^{\circ}\text{F}$ .

### 11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considering an environmental temperature of  $100^{\circ}\text{F}$  for a duration sufficient to reach thermal equilibrium is evaluated with respect to design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The  $100^{\circ}\text{F}$  environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Subsection 4.4. These temperatures are conservatively calculated at an environmental temperature of  $80^{\circ}\text{F}$ . The maximum off-normal environmental temperature is  $100^{\circ}\text{F}$ , which is an increase of  $20^{\circ}\text{F}$ . ~~Including the effect of a hypothetical 10% rod rupture condition on the MPC cavity gas conductivity, including this conservatively as a bounding temperatures increment for all MPC designs (Table 1.2.1) of the MPC-68 and MPC-24 over the  $80^{\circ}\text{F}$  ambient temperature solutions of Chapter 4, are calculated to be as listed in~~ the HI-STORM temperatures are computed and provided in Table 11.1.1. As illustrated by the table, all the maximum off-normal temperatures are below the ~~short-term condition~~ off-normal design basis temperatures for the HI-STORM System (Table 2.2.3). The maximum temperatures are the peak values and are based on the conservative assumptions applied in this analysis. The component temperatures for the HI-TRAC listed in Table 4.5.2 are all based on the maximum off-normal environmental temperature. The off-normal environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term temperature limits. Therefore, all the HI-STORM 100 System maximum off-normal temperatures meet the design requirements.

Additionally, the off-normal environmental temperature generates a pressure that is *bounded by that* evaluated in Subsection 11.1.1. The off-normal MPC cavity pressure is less than the design basis pressure listed in Table 2.2.1.

The off-normal event considering an environmental temperature of  $-40^{\circ}\text{F}$  and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM overpack. The HI-STORM overpack and MPC are conservatively assumed to reach  $-40^{\circ}\text{F}$  throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is  $0^{\circ}\text{F}$  and the HI-TRAC is conservatively assumed to reach  $0^{\circ}\text{F}$  throughout the structure. For ambient temperatures from  $0^{\circ}$  to  $32^{\circ}\text{F}$ , a 25%

~~ethylene glycol solution is an~~ *antifreeze must be* added to the demineralized water in the water jacket to prevent freezing. Chapter 3, Subsection 3.1.2.3, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100 System.

### Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 11.1.1.3, the resultant pressure is ~~well~~ below the *off-normal* design pressure (*Table 2.2.1 in Chapter 2*) of 100 psig used in the structural analysis. The effect of the lower off-normal thermal conditions (i.e., -40°F) ~~results in~~ *requires* an evaluation of the potential for brittle fracture. ~~That~~ *Such an evaluation is discussed* presented in Section 3.1.2.3.

### Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Table 11.1.1 for the HI-STORM overpack and MPC. As can be seen from this table, all temperatures for off-normal conditions are within the short-term allowable values ~~described~~ *listed* in Table 2.2.3.

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100 System.

#### 11.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no corrective actions required for off-normal environmental temperatures.

#### 11.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

#### 11.1.3 Leakage of One Seal

The HI-STORM 100 System has a reliable welded boundary to contain radioactive fission products within the confinement boundary. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, and vent and drain port cover plates. The closure ring provides a redundant welded closure to the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. ~~The welds on the MPC lid, and vent and drain port covers are leakage tested.~~ Additionally, the MPC lid weld is subjected to a hydrostatic test to verify its integrity.

*Section 7.1 provides a discussion as to how the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.*

~~The MPC lid to MPC shell weld is postulated to fail to confirm the safety of the HI-STORM 100 confinement boundary. The failure of the MPC lid weld is equivalent to the MPC drain or vent port cover weld failing. The MPC lid to shell weld has been selected because it is the main closure weld performed in the field for the MPC. It is extremely unlikely that the weld examination, helium leakage testing and hydrostatic testing would fail to detect a poorly welded closure plate. The MPC lid weld failure affects the MPC confinement boundary; however, no leakage will occur.~~

#### 11.1.3.1 Postulated Cause of Leakage of One Seal in the Confinement Boundary

*There is no credible cause for the leakage of one seal in the confinement boundary. The conditions analyzed in Chapter 3 shows that the confinement boundary components are maintained within their Code-allowable stress limits under all normal and off-normal storage conditions. The MPC fabrication and closure welds meet the requirements of ISG-18, such that leakage from the confinement boundary is not considered credible. Therefore, there is no event that could cause leakage of one seal in the confinement boundary.*

~~Failure of the MPC confinement boundary is highly unlikely. The MPC confinement boundary is shown to withstand all normal, off-normal, and accident conditions. There are no credible conditions that could damage the integrity of the MPC confinement boundary. The MPC lid to MPC shell weld is liquid penetrant inspected on the root and final pass, volumetrically inspected or liquid penetrant inspected on multiple passes, hydrostatically tested, and helium leak tested. The initial integrity of the closure welds will be maintained throughout the design life because the MPC is stored within the~~

HI-STORM overpack which provides physical protection and a weather shield. Failure of the MPC lid to MPC shell weld would require all of the following:

1. ~~Improper weld by a qualified welding machine or welder using approved welding procedures.~~
2. ~~Failure to detect the unacceptable indication during the liquid penetrant or volumetric inspections performed by a qualified inspector in accordance with approved procedures.~~
3. ~~Failure of the qualified leakage test equipment to detect the leak in accordance with approved procedures.~~
4. ~~Failure to detect the unacceptable leak during the hydrostatic test performed by qualified personnel in accordance with approved procedures.~~

The evaluation of the failure of the MPC lid to MPC shell weld has been postulated to demonstrate the safety of the HI-STORM 100 confinement system and cannot be derived from a credible loading condition.

#### 11.1.3.2 Detection of Leakage of One Seal in the Confinement Boundary

The HI-STORM 100 System is designed ~~to withstand the~~ *such that* leakage of one field weld in the confinement boundary ~~without any effects on its ability to meet its safety requirements~~ *is not considered a credible scenario*. ~~As the HI-STORM 100 System can withstand the failure of one field weld with no leakage~~ *Therefore*, there is no requirement to detect leakage from one seal.

#### 11.1.3.3 Analysis of Effects and Consequences of Leakage of One Seal in the Confinement Boundary

If the MPC lid to MPC shell weld were to fail, the MPC closure ring will retain the design pressure. The analysis of the MPC closure ring's ability to retain the design pressure is provided in Appendix 3.E of the HI-STAR TSAR Docket Number 72-1008. The consequences of the MPC lid to MPC shell weld failure are that the MPC closure ring maintains the integrity of the confinement boundary.

#### Structural

The stress evaluation of the closure ring is discussed in Appendix 3.E of the HI-STAR TSAR Docket Number 72-1008. All stresses are within the allowable values.

#### Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

#### Shielding

~~There is no effect on the shielding performance of the system as a result of this off-normal event.~~

#### Criticality

~~There is no effect on the criticality control features of the system as a result of this off-normal event.~~

#### Confinement

~~There is no effect on the confinement function of the MPC as a result of this off-normal event.~~

#### Radiation Protection

~~Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.~~

~~Based on this evaluation, it is concluded that the specified off-normal leakage of one seal event does not affect the safe operation of the HI-STORM 100 System.~~

#### 11.1.3.34 Corrective Action for Leakage of One Seal in the Confinement Boundary

There is no corrective action required for the failure of one weld in the closure system of the confinement boundary. Leakage of one weld in the confinement boundary closure system ~~does not affect the HI-STORM 100 System's ability to operate safely~~ *is not a credible event.*

#### 11.1.3.45 Radiological Impact of Leakage of One Seal in the Confinement Boundary

The off-normal event of the failure of one weld in the confinement boundary closure system has no radiological impact because ~~the leakage from the confinement barrier is not breached and shielding is not affected~~ *credible.*

#### 11.1.4 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with fine mesh screens on the inlet and outlet air ducts. These screens ensure the air ducts are protected from the incursion of foreign objects. There are four air inlet ducts 90° apart and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet ducts. As required by the design criteria presented in Chapter 2, it is conservatively assumed that two of the four air inlet ducts are blocked. The blocked air inlet ducts are assumed to be completely blocked with an ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the inherent thermal stability of the HI-STORM 100 System.

~~An additional evaluation is performed with three of the four air inlet ducts. While not required by the HI-STORM System design criteria, this additional evaluation is performed as a parametric study of the effects of incremental duct blockage. The purpose of the parametric study is to demonstrate the robustness of the HI-STORM System design beyond the design basis.~~



#### 11.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is conservatively assumed that the blocked air inlet ducts are completely blocked, although mesh screens prevent foreign objects from entering the ducts. The mesh screens are either inspected periodically or the outlet duct air temperature is monitored as specified by Technical Specifications in Appendix A to the CoC. It is, however, possible that blowing debris may block two air inlet ducts of the overpack. As already stated, the blockage of three inlet ducts is evaluated only to demonstrate the limited effects of additional incremental duct blockage.

#### 11.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet ducts will occur during the routine visual inspection of the mesh screens or temperature monitoring of the outlet duct air as required and specified by Technical Specifications in Appendix A to the CoC. The frequency of inspection is based on an assumed complete blockage of all four air inlet ducts. There is no inspection requirement as a result of the postulated two inlet duct blockage, because the complete blockage of all four air inlet ducts is bounding.

#### 11.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

~~Evaluations for~~The two inlet ducts and three inlet ducts blocked are condition is evaluated for the hottest MPC-68-MPC-32 at its maximum decay heat load. Only the MPC-32 is evaluated because it has the highest decay heat load of all MPC designs (Table 1.2.1). The largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is 25°F, for the MPC shell. The largest temperature rise of the MPC or its contents as a result of the blockage of three air inlet ducts (performed as a parametric study of incremental duct blockage only) is 81°F, also for the MPC shell. Conservatively adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of two air inlet ducts are provided in Table 11.1.2. ~~Following this same procedure of adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of three air inlet ducts are included in the same table for comparison purposes. These values are based on full insolation and an ambient temperature of 80°F. The analysis method for the blockage of two and three of the air inlet ducts is conservative with respect to the analysis method for the normal condition. As a result of the air inlet duct blockages, the head loss is increased and the airflow is decreased thereby increasing component temperatures.~~

As stated above, the largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is 25°F, for the MPC shell. A bounding MPC internal pressure as a result of this calculated temperature increase is computed, based on initial conditions presented previously in Subsection 11.1.1.3, as follows:

$$P_2 = P_1 \frac{T_1 + \Delta T}{T_1}$$

where:

- $P_2$  = Bounding MPC Cavity Pressure (psia)
- $P_1$  = Initial MPC Cavity Pressure (89.7 psia)
- $T_1$  = Initial MPC Cavity Average Temperature (513.6°K)
- $\Delta T$  = Bounding MPC Temperature Rise (25°F or 193.9°K)

Substituting these values into the equation above, the bounding MPC internal pressure is obtained as:

$$P_2 = 89.7 \times \frac{513.6 + 13.9}{513.6} = 92.1 \text{ psia} = 77.4 \text{ psig}$$

~~The off-normal MPC internal design pressure of 100 psig (Table 2.2.1) has been established to bounds this partial inlet duct blockage condition.~~

~~Although it is a beyond the design basis condition, the bounding pressure rise for the three blocked air inlet ducts condition can be determine in the same manner. As stated above, the bounding temperature rise for this condition is 81°F (44.9°K), and the corresponding bounding MPC internal pressure is 97.5 psia (82.8 psig). This parametric evaluation demonstrates the insensitivity of the MPC internal pressure to incremental duct blockage, as the relatively large incremental flow area reduction increases the pressure by only 5.4 psi.~~

### Structural

There are no structural consequences as a result of this off-normal event.

### Thermal

Using the methodology and model discussed in Section 4.4, the thermal analysis for the two air inlet ducts blocked off-normal condition is performed. The analysis demonstrates that under steady-state conditions, no system components exceed the short-term allowable temperatures in Table 2.2.3.

~~The parametric study of incremental duct blockage, performed by evaluating a three air inlet ducts blocked condition, demonstrates the insensitivity of the system to relatively large incremental flow area reductions. This beyond the design basis condition results in relatively small temperature increases and temperatures well below the short-term allowable temperatures in Table 2.2.3, even though no such requirement exists.~~

### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

## Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100 System.

### 11.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet ducts does not affect the HI-STORM 100 System's ability to operate safely.

*Periodic inspection* of the HI-STORM overpack air duct screen covers is required ~~with the frequency specified by Technical Specifications in Appendix A to the CoC.~~ Alternatively, the outlet duct air temperature is monitored. The frequency of inspection is based on an assumed blockage of all four air inlet ducts analyzed in Subsection 11.2.

### 11.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet ducts has no radiological impact because the confinement barrier is not breached and shielding is not affected.

### 11.1.5 Off-Normal Handling of HI-TRAC

During upending and/or downending of the HI-TRAC transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation.

If the lifting device is allowed to "go slack", the total weight would be applied to the lower pocket trunnions only. Under this off-normal condition, the pocket trunnions would each be required to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to

demonstrate that the pocket trunnions possess sufficient strength to support the increased load under this off-normal condition.

This off-normal condition does not apply to the HI-TRAC 125D, which does not have lower pocket trunnions. Upending and downending of the HI-TRAC 125D is performed using an L-frame.

#### 11.1.5.1 Postulated Cause of Off-Normal Handling of HI-TRAC

If the cable of the crane handling the HI-TRAC is inclined from the vertical, it would be possible to unload the upper lifting trunnions such that the lower pocket trunnions are supporting the total cask weight and the lifting trunnions are only preventing cask rotation.

#### 11.1.5.2 Detection of Off-Normal Handling of HI-TRAC

Handling procedures and standard rigging practice call for maintaining the crane cable in a vertical position by keeping the crane trolley centered over the lifting trunnions. In such an orientation it is not possible to completely unload the lifting trunnions without inducing rotation. If the crane cable were inclined from the vertical, however, the possibility of unloading the lifting trunnions would exist. It is therefore possible to detect the potential for this off-normal condition by monitoring the incline of the crane cable with respect to the vertical.

#### 11.1.5.3 Analysis of Effects and Consequences of Off-Normal Handling of HI-TRAC

If the upper lifting trunnions are unloaded, the lower pocket trunnions will support the total weight of the loaded HI-TRAC. The analysis of the pocket trunnions to support the applied load of one-half of the total weight is provided in ~~Appendices 3-AA and 3-AI~~ *Subsection 3.4.4.3.3.1* of this FSAR. The consequence of off-normal handling of the HI-TRAC is that the pocket trunnions safely support the applied load.

#### Structural

The stress evaluations of the lower pocket trunnions are discussed in *Subsection 3.4.4.3.3.1 of this FSAR* ~~Appendices 3-AA and 3-AI~~. All stresses are within the allowable values.

#### Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

## Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal handling of the HI-TRAC does not affect the safe operation of the system.

### 11.1.5.4 Corrective Action for Off-Normal Handling of HI-TRAC

The HI-TRAC transfer casks are designed to withstand the off-normal handling condition without any adverse effects. There are no corrective actions required for off-normal handling of HI-TRAC other than to attempt to maintain the crane cable vertical during HI-TRAC upending or downending.

### 11.1.5.5 Radiological Consequences of Off-Normal Handling of HI-TRAC

The off-normal event of off-normal handling of HI-TRAC has no radiological impact because the confinement barrier is not breached and shielding is not affected.

### 11.1.6 Failure of FHD System Off-Normal Load Combinations

~~Load combinations for off-normal conditions are provided in Table 2.2.14. The load combinations include normal loads with the off-normal loads. The load combination results are shown in Section 3.4 to meet all allowable values.~~

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

#### 11.1.6.1 Postulated Cause of FHD Failure

*Likely causes of FHD failure are (i) Loss of external power to the FHD System and (ii) An active component trip. In both cases a stoppage of forced helium circulation occurs.*

### 11.1.6.2 Detection of FHD Failure

*The HI-STORM 100 System is designed to withstand an FHD System failure without affecting its ability to meet safety requirements. Consequently FHD monitoring and failure detection is not required.*

### 11.1.6.3 Analysis of Effects and Consequences of FHD Failure

#### Structural

*The FHD System is required to be equipped with safety relief devices§ to prevent the MPC structural boundary pressures from exceeding the design limits. Consequently there is no adverse effect.*

#### Thermal

*Failure of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit is 1058°F (Table 2.2.3). The FHD System failure event is evaluated assuming the following bounding conditions:*

- 1) Steady state maximum temperatures have been reached*
- 2) Design basis heat load*
- 3) Standing column of air in the annulus*
- 4) MPCs backfilled with the minimum helium pressure required by the Technical Specifications*

*The steady state results are provided in Table 11.1.3. The results demonstrate that the peak fuel cladding temperatures remain below the limit in the event of a prolonged unavailability of the FHD system.*

#### Shielding

*There is no effect on the shielding performance of the system as a result of this off-normal event.*

#### Criticality

*There is no effect on the criticality control of the system as a result of this off-normal event.*

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*§ The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 5 atm to enable MPC pressurization for adequate heat transfer.*

### Confinement

*There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, the structural boundary pressures cannot exceed the design limits.*

### Radiation Protection

*As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.*

*Based on this evaluation, it is concluded that the FHD failure does not affect the safe operation of the HI-STORM 100 System.*

#### 11.1.6.4 Corrective Action for FHD Failure

*The HI-STORM 100 System is designed to withstand the FHD failure without an adverse effect on its safety functions. Consequently no corrective action is required.*

#### 11.1.6.5 Radiological Impact of FHD Failure

*The event has no radiological impact because the confinement barrier and shielding integrity are not affected.*

#### 11.1.7 SCS Power Failure

*The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation.*

##### 11.1.7.1 Postulated Cause of SCS Power Failure

*The SCS is normally operated from an external source of power such as from site utilities or a feed from a heavy haul vehicle carrying the HI-TRAC. Occasional interruption in power supply is possible.*

##### 11.1.7.2 Detection of SCS Power Failure

*The HI-STORM 100 System is designed to withstand a power failure without affecting its ability to meet safety requirements. Consequently SCS monitoring and failure detection is not required.*

### *11.1.7.3      Analysis of Effects and Consequences of SCS Power Failure*

*The SCS System is required to be equipped with a backup power supply (See SCS specifications in Chapter 2, Appendix 2.C). This ensures uninterrupted operation of the SCS following a power failure. Consequently, a power failure does not effect SCS operation.*

#### *Structural*

*There is no effect on the structural integrity.*

#### *Thermal*

*There is no effect on thermal performance.*

#### *Shielding*

*There is no effect on the shielding performance.*

#### *Criticality*

*There is no effect on the criticality control.*

#### *Confinement*

*There is no effect on the confinement function.*

#### *Radiation Protection*

*As there is no effect on the shielding or confinement functions, there is no effect on occupational or public exposures.*

*Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.*

### *11.1.7.4      Corrective Action for SCS Power Failure*

*The HI-STORM 100 System is designed to withstand a power failure without an adverse effect on its normal operation. Consequently no corrective action is required.*

### *11.1.7.5      Radiological Impact of SCS Power Failure*

*The event has no radiological impact because the confinement barrier and shielding integrity are not affected.*



Table 11.1.1

MAXIMUM TEMPERATURES CAUSED BY OFF-NORMAL ENVIRONMENTAL TEMPERATURES

<b>Location</b>	<b>Temperature [°F]</b>	<b>Design Basis Limits [°F]</b>
Fuel Cladding	711 (PWR) 760 (BWR)	<del>1058 short term</del>
MPC Basket	740	<del>950 short term</del>
MPC Shell	371	<del>775 short term</del>
<del>Overpack Air Outlet</del>	<del>226</del>	<del>N/A</del>
Overpack Inner Shell	219	<del>350 short term (overpack concrete)</del>
Overpack Outer Shell	165	<del>350 short term (overpack concrete)</del>

Table 11.1.2

*BOUNDING*<sup>‡</sup> TEMPERATURES CAUSED BY PARTIAL BLOCKAGE OF AIR INLET DUCTS [°F]

Temperature Location	No Blockage of Inlet Ducts	Partial Blockage of Inlet Ducts		Off-Normal Design-Basis
		2 Ducts Blocked	3 Ducts Blocked	
Fuel Cladding	740	765	821	<del>1058 short-term</del>
MPC Basket	720	745	801	<del>950 short-term</del>
MPC Shell	351	376	432	<del>775 short-term</del>
Overpack Air Outlet	206	231	287	N/A
Overpack Inner Shell	199	224	280	350 short-term (overpack concrete)
Overpack Outer Shell	145	170	226	350 short-term (overpack concrete)

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<sup>‡</sup>- *The bounding temperatures presented in this table are obtained by adding the maximum temperature rise of any cask component to the normal condition temperatures of every cask component.*

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Table 11.1.3

*BOUNDING STEADY-STATE FUEL CLADDING TEMPERATURES  
FOLLOWING AN FHD FAILURE*

<i>MPC</i>	<i>Computed Peak Clad Temp. (°F)</i>	<i>Off-Normal Temperature Limit (°F)</i>
<i>All</i>	872	1058

## 11.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM 100 System or events postulated because their consequences may affect the public health and safety. Section 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM 100 System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. The following sections present the evaluation of the design basis postulated accident conditions and natural phenomena which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for postulated accident conditions are defined in Table 2.2.14. The load combinations include normal loads with the accident loads. The accident load combination evaluations are provided in Section 3.4.

### 11.2.1 HI-TRAC Transfer Cask Handling Accident

#### 11.2.1.1 Cause of HI-TRAC Transfer Cask Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-TRAC transfer cask can be transported to the ISFSI in the vertical or horizontal position. The loaded HI-TRAC transfer cask is typically transported by a heavy-haul vehicle that cradles the HI-TRAC horizontally or by a device with redundant drop protection that holds the HI-TRAC vertically. The height of the loaded overpack above the ground shall be limited to below the horizontal handling height limit determined in Chapter 3 and specified by the Technical Specifications in Appendix A to the CoC to limit the inertia loading on the cask in a horizontal drop to less than 45g's. Although a handling accident is remote, a cask drop from the horizontal handling height limit is a credible accident. A vertical drop of the loaded HI-TRAC transfer cask is not a credible accident as the loaded HI-TRAC shall be transported and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3.1 as required by the Technical Specification.

#### 11.2.1.2 HI-TRAC Transfer Cask Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded HI-TRAC in the horizontal position. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the ~~HI-STORM 100 System~~ HI-TRAC meets all structural requirements and there is no adverse effect on the confinement, thermal or subcriticality performance of the contained MPC. Limited localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket may occur as a result of the handling accident. The HI-TRAC top lid and transfer lid housing (pool lid for the HI-TRAC 125D) are demonstrated to remain attached by withstanding the

maximum deceleration. The transfer lid doors (not applicable to HI-TRAC 125D) are also shown to remain closed during the drop. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1]. Therefore, demonstrating that the 45g limit for the HI-TRAC transfer cask is met ensures that the fuel cladding remains intact.

### Structural

The structural evaluation of the MPC for 45g's is provided in Section 3.4. As discussed in Section 3.4, the MPC stresses as a result of the HI-TRAC side drop, 45g loading, are all within allowable values.

As discussed above, the water jacket enclosure shell could be punctured which results in a loss of the water within the water jacket. Additionally, the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) are shown to remain in position under the 45g loading. Analysis of the lead in the HI-TRAC is performed in Appendix 3.F and it is shown that there is no appreciable change in the lead shielding.

### Thermal

The loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

### Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded. As the structural analysis demonstrates that the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) remain in place, there is no change in the dose rates at the top and bottom of the HI-TRAC.

### Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

## Radiation Protection

There is no degradation in the confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. The dose rate at 1 meter from the water jacket after the water is lost is calculated in Table 5.1.10. Immediately after the drop accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public. Based on a minimum distance to the controlled area boundary of 100 meters, the *10CFR72.106* dose rate requirements at the controlled area boundary (*5 Rem limit*) will be approximately 1.48 mrem/hr (Section 5.1.2). Therefore, it is evident, based on the short duration of the accident, that the requirements of *10CFR72.106* (*5 Rem*) will not be exceeded (Section 5.1.2).

### 11.2.1.3 HI-TRAC Transfer Cask Handling Accident Dose Calculations

The handling accident could cause localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket as the neutron shield impacts the ground.

When the water jacket is impacted, the HI-TRAC transfer cask surface dose rate could increase. The HI-TRAC's post-accident shielding analysis presented in Section 5.1.2 assumes complete loss of the water in the water jacket and bounds the dose rates anticipated for the handling accident.

If the water jacket of the loaded HI-TRAC is damaged beyond immediate repair and the MPC is not damaged, the loaded HI-TRAC may be unloaded into a HI-STORM overpack, a HI-STAR overpack, or simply unloaded in the fuel pool. If the MPC is damaged, the loaded HI-TRAC must be returned to the fuel pool for unloading. Depending on the damage to the HI-TRAC and the current location in the loading or unloading sequence, less personnel exposure may be received by continuing to load the MPC into a HI-STORM or HI-STAR overpack. Once the MPC is placed in the HI-STORM or HI-STAR overpack, the dose rates are greatly reduced. The highest personnel exposure will result from returning the loaded HI-TRAC to the fuel pool to unload the MPC.

As a result of the loss of water from the water jacket, the dose rates at 1 meter adjacent to the water jacket mid-height increase (Table 5.1.10). Increasing the personnel exposure for each task affected by the increased dose rate adjacent to the water jacket by the ratio of the one meter dose rate increase results in a cumulative dose of less than 15.0 person-rem, for the 125-ton HI-TRAC or 100-ton HI-TRAC. Using the ratio of the water jacket mid-height dose rates at one meter is very conservative. Dose rate at the top and bottom of the HI-TRAC water jacket would not increase as much as the peak mid-height dose rates. In the determination of the personnel exposure, dose rates at the top and bottom of the loaded HI-TRAC are assumed to remain constant.

The analysis of the handling accident presented in Section 3.4 shows that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactive material from the confinement vessel. Any possible rupture of the fuel cladding will have no effect on the site boundary dose rates because the magnitude of the radiation source has not changed.

#### 11.2.1.4 HI-TRAC Transfer Cask Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the HI-TRAC transfer cask and MPC to the maximum practical extent. As appropriate, place temporary shielding around the HI-TRAC to reduce radiation dose rates. Special handling procedures will be developed and approved by the ISFSI operator to lift and upright the HI-TRAC. Upon uprighting, the portion of the overpack not previously accessible shall be radiologically and visually inspected. If damage to the water jacket is limited to a local penetration or crushing, local repairs can be performed to the shell and the water replaced. If damage to the water jacket is extensive, the damage shall be repaired and re-tested in accordance with Chapter 9, following removal of the MPC.

If upon inspection of the damaged HI-TRAC transfer cask and MPC, damage of the MPC is observed, the loaded HI-TRAC transfer cask will be returned to the facility for fuel unloading in accordance with Chapter 8. The handling accident will not affect the ability to unload the MPC using normal means as the structural analysis of the 60g loading (HI-STAR Docket Numbers 71-9261 and 72-1008) shows that there will be no gross deformation of the MPC basket. After unloading, the structural damage of the HI-TRAC and MPC shall be assessed and a determination shall be made if repairs will enable the equipment to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the equipment for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

#### 11.2.2 HI-STORM Overpack Handling Accident

##### 11.2.2.1 Cause of HI-STORM Overpack Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-STORM overpack is lifted in the vertical orientation. The height of the loaded overpack above the ground shall be limited to below the vertical handling height limit determined in Chapter 3 ~~and specified by the Technical Specifications in Appendix A to the CoC~~. This vertical handling height limit will maintain the inertial loading on the cask in a vertical drop to 45g's or less. Although a handling accident is remote, a drop from the vertical handling height limit is a credible accident.

##### 11.2.2.2 HI-STORM Overpack Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded overpack in the vertical orientation. The analysis of the handling accident is provided in Chapter 3. The analysis shows that

the HI-STORM 100 System meets all structural requirements and there are no adverse effects on the structural, confinement, thermal or subcriticality performance of the HI-STORM 100 System. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1].

### Structural

The structural evaluation of the MPC under a 60g vertical load is presented in the HI-STAR TSAR and SAR [11.2.6 and 11.2.7] and it is demonstrated therein that the stresses are within allowable limits. The structural analysis of the HI-STORM overpack is presented in Section 3.4. The structural analysis of the overpack shows that the concrete shield attached to the underside of the overpack lid remains attached and air inlet ducts do not collapse.

### Thermal

As the structural analysis demonstrates that there is no change in the MPC or overpack, there is no effect on the thermal performance of the system as a result of this event.

### Shielding

As the structural analysis demonstrates that there is no change in the MPC or overpack, there is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the vertical drop of the HI-STORM Overpack with the MPC inside does not affect the safe operation of the HI-STORM 100 System.



### 11.2.2.3 HI-STORM Overpack Handling Accident Dose Calculations

The vertical drop handling accident of the loaded HI-STORM overpack will not cause any change of the shielding or breach of the MPC confinement boundary. Any possible rupture of the fuel cladding will have no effect on the site boundary dose rates because the magnitude of the radiation source has not changed. Therefore, the dose calculations are equivalent to the normal condition dose rates.

### 11.2.2.4 HI-STORM Overpack Handling Accident Corrective Action

Following a handling 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, as required, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC is to be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs 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.

### 11.2.3 Tip-Over

#### 11.2.3.1 Cause of Tip-Over

The analysis of the HI-STORM 100 System has shown that the overpack does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. It is highly unlikely that the overpack will tip-over during on-site movement because of the low handling height limit. The tip-over accident is stipulated as a non-mechanistic accident.

For the anchored HI-STORM designs (HI-STORM 100A and 100SA), a tip-over accident is not possible. As described in Chapter 2 of this FSAR, these system designs are not evaluated for the hypothetical tip-over. As such, the remainder of this accident discussion applies only to the non-anchored designs (i.e., the 100 and 100S designs only).

#### 11.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Section 3.4. The structural analysis provided in Appendix 3.A demonstrates that the resultant deceleration loading on the MPC as a result of the tip-over accident is less than the design basis 45g's. The analysis shows that the HI-STORM 100 System meets all structural requirements and there is no adverse effect on the structural, confinement, thermal, or subcriticality performance of the MPC. However, the side

impact will cause some localized damage to the concrete and outer shell of the overpack in the radial area of impact.

### Structural

The structural evaluation of the MPC presented in Section 3.4 demonstrates that under a 45g loading the stresses are well within the allowable values. Analysis presented in Chapter 3 shows that the concrete shields attached to the underside and top of the overpack lid remains attached. As a result of the tip-over accident there will be localized crushing of the concrete in the area of impact.

### Thermal

The thermal analysis of the overpack and MPC is based on vertical storage. The thermal consequences of this accident while the overpack is in the horizontal orientation are bounded by the burial under debris accident evaluated in Subsection 11.2.14. Damage to the overpack will be limited as discussed above. As the structural analysis demonstrates that there is no significant change in the MPC or overpack, once the overpack and MPC are returned to their vertical orientation there is no effect on the thermal performance of the system.

### Shielding

The effect on the shielding performance of the system as a result of this event is limited to a localized decrease in the shielding thickness of the concrete.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the accident pressure does not affect the safe operation of the HI-STORM 100 System.

### 11.2.3.3 Tip-Over Dose Calculations

The tip-over accident could cause localized damage to the radial concrete shield and outer steel shell where the overpack impacts the surface. The overpack surface dose rate in the affected area could increase due to the damage. However, there should be no noticeable increase in the ISFSI site or boundary dose rate, because the affected areas will be small and localized. The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates.

### 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 will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC shall be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. 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 due to the lack of combustible materials at the ISFSI, 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.

### 11.2.4.2 Fire Analysis

#### 11.2.4.2.1 Fire Analysis for HI-STORM Overpack

The possibility of a fire accident near an ISFSI is considered to be extremely remote due to an absence of combustible materials within the ISFSI and adjacent to the overpacks. The only credible concern is related to a transport vehicle fuel tank fire, causing the outer layers of the storage overpack to be heated by the incident thermal radiation and forced convection heat fluxes. The amount of combustible fuel in the on-site transporter is limited to a volume of 50 gallons based on a ~~Technical Specification in Appendix A to the CoC.~~

With respect to fire accident thermal analysis, NUREG-1536 (4.0,V,5.b) states:

“Fire parameters included in 10 CFR 71.73 have been accepted for characterizing the heat transfer during the in-storage fire. However, a bounding analysis that limits the fuel source thus limits the length of the fire (e.g., by limiting the source of the fuel in the transporter) has also been accepted.”

Based on this NUREG-1536 guidance, the fire accident thermal analysis is performed using the 10 CFR 71.73 parameters and the fire duration is determined from the limited fuel volume of 50 gallons. The entire transient evaluation of the storage fire accident consists of three parts: (1) a bounding steady-state initial condition, (2) the short-duration fire event, and (3) the post-fire temperature relaxation period.

As stated above, the fire parameters from 10 CFR 71.73 are applied to the HI-STORM fire accident evaluation. 10 CFR 71 requirements for thermal evaluation of hypothetical accident conditions specifically define pre- and post-fire ambient conditions, specifically:

“the ambient air temperature before and after the test must remain constant at that value between -29°C (-20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration.”

The ambient air temperature is therefore set to 100°F both before (bounding steady state) and after (post-fire temperature relaxation period) the short-duration fire event.

During the short-duration fire event, the following parameters from 10CFR71.71(c)(4), *also from Reference [11.2.3]*, are applied:

1. Except for a simple support system, the cask must be fully engulfed. The ISFSI pad is a simple support system, so the fire environment is not applied to the overpack baseplate. By fully engulfing the overpack, additional heat transfer surface area is conservatively exposed to the elevated fire temperatures.
2. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.
3. The average flame temperature must be at least 800°C (1475°F). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces,

which is very conservative considering the size of the HI-STORM cask. It is therefore conservative to use the 1475°F temperature.

4. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration.
5. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal measurements [11.2.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft<sup>2</sup>×°F) is applied to exposed overpack surfaces during the short-duration fire.

Due to the severity of the fire condition radiative heat flux, heat flux from incident solar radiation is negligible and is not included. Furthermore, the smoke plume from the fire would block most of the solar radiation.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width, the fuel ring surrounding the overpack covers 147.6 ft<sup>2</sup> and has a depth of 0.54 in. From this depth and a linear fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.622 minutes (217 seconds). The linear fuel consumption rate of 0.15 in/min is the smallest value given in a Sandia Report on large pool fire thermal testing [11.2.2]. Use of the minimum linear consumption rate conservatively maximizes the duration of the fire.

It is recognized that the ventilation air in contact with the inner surface of the HI-STORM overpack with design-basis decay heat under maximum normal ambient temperature conditions varies between 80°F at the bottom and 206°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy only 1.25% of area of the cylindrical surface of the massive HI-STORM overpack. Due to the short duration of the fire event and the relative isolation of the ventilation passages from the outside environment, the ventilation air is expected to experience little intrusion of the fire combustion products. As a result of these considerations, it is conservative to assume that the air in the HI-STORM overpack ventilation passages is held constant at a substantially elevated temperature of 300°F during the entire duration of the fire event.

The thermal transient response of the storage overpack is determined using the ANSYS finite element program. Time-histories for points in the storage overpack are monitored for the duration of the fire and the subsequent post-fire equilibrium phase.

Heat input to the HI-STORM overpack while it is subjected to the fire is from a combination of an incident radiation and convective heat fluxes to all external surfaces. This can be expressed by the following equation:

$$q_F = h_{fc} (T_A - T_S) + 0.1714 \times 10^8 \varepsilon [(T_A + 460)^4 - (T_S + 460)^4]$$

where:

- $q_F$  = Surface Heat Input Flux (Btu/ft<sup>2</sup>-hr)
- $h_{fc}$  = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft<sup>2</sup>-hr-°F)
- $T_A$  = Fire Condition Temperature (1475°F)
- $T_S$  = Transient Surface Temperature (°F)
- $\varepsilon$  = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [11.2.2].

After the fire event, the ambient temperature is restored to 100°F and the storage overpack cools down (post-fire temperature relaxation). Heat loss from the outer surfaces of the storage overpack is determined by the following equation:

$$q_S = h_S (T_S - T_A) + 0.1714 \times 10^8 \varepsilon [(T_S + 460)^4 - (T_A + 460)^4]$$

where:

- $q_S$  = Surface Heat Loss Flux (Btu/ft<sup>2</sup>-hr)
- $h_S$  = Natural Convection Heat Transfer Coefficient (Btu/ft<sup>2</sup>-hr-°F)
- $T_S$  = Transient Surface Temperature (°F)
- $T_A$  = Ambient Temperature (°F)
- $\varepsilon$  = Surface Emissivity

In the post-fire temperature relaxation phase, the surface heat transfer coefficient ( $h_S$ ) is determined by the following equation:

$$h_S = 0.19 \times (T_A - T_S)^{1/3}$$

where:

- $h_S$  = Natural Convection Heat Transfer Coefficient (Btu/ft<sup>2</sup>-hr-°F)
- $T_A$  = External Air Temperature (°F)
- $T_S$  = Transient Surface Temperature (°F)

As discussed in Subsection 4.5.1.1.2, this equation is appropriate for turbulent natural convection from vertical surfaces. For the same conservative value of the Z parameter assumed earlier ( $2.6 \times 10^5$ ) and the HI-STORM overpack height of approximately 19 feet, the surface-to-ambient temperature difference required to ensure turbulence is 0.56 °F.

A two-dimensional, axisymmetric model was developed for this analysis. Material thermal properties used were taken from Section 4.2. An element plot of the 2-D axisymmetric ANSYS model is shown in Figure 11.2.1. The outer surface and top surface of the overpack are exposed to the ambient conditions (fire and post-fire), and the base of the overpack is insulated. The transient

study is conducted for a period of 5 hours, which is sufficient to allow temperatures in the overpack to reach their maximum values and begin to recede.

Based on the results of the analysis, the maximum temperatures ~~increases~~ at several points near the overpack mid-height are summarized in Table 11.2.2 along with the corresponding peak temperatures *in the MPC*. ~~Temperature profiles through the storage overpack wall thickness near the mid-height of the cask are included in Figures 11.2.2 through 11.2.4. A plot of temperature versus time is shown in Figure 11.2.5 for several points through the overpack wall, near the mid-height of the cask. The temperature profile plots (Figures 11.2.2 through 11.2.4) each contain profiles corresponding to time “snapshots”. Profiles are presented at the following times: 1 minute (60 seconds), 2 minutes (120 seconds), 3.622 minutes (217 seconds — end of fire), 10 minutes (600 seconds), 20 minutes (1200 seconds), 40 minutes and 90 minutes.~~

The primary shielding material in the storage overpack is concrete, which can suffer a reduction in neutron shielding capability at sustained high temperatures due to a loss of water. ~~As shown in Figure 11.2.5,~~ Less than 1 inch of the concrete near the outer overpack surface exceeds the material short-term temperature limit. This condition is addressed specifically in NUREG-1536 (4.0,V,5.b), which states:

“The NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire.”

These results demonstrate that the fire accident event does not substantially affect the HI-STORM overpack. Only localized regions of concrete are exposed to temperatures in excess of the allowable short-term temperature limit. No portions of the steel structure exceed the allowable temperature limits.

Having evaluated the effects of the fire on the overpack, we must now evaluate the effects on the MPC and contained fuel assemblies. Guidance for the evaluation of the MPC and its internals during a fire event is provided by NUREG-1536 (4.0,V,5.b), which states:

“For a fire of very short duration (i.e., less than 10 percent of the thermal time constant of the cask body), the NRC finds it acceptable to calculate the fuel temperature increase by assuming that the cask inner wall is adiabatic. The fuel temperature increase should then be determined by dividing the decay energy released during the fire by the thermal capacity of the basket-fuel assembly combination.”

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)
- $\rho$  = Overpack Density (lb/ft<sup>3</sup>)
- $L_c$  = 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 (142 lb/ft<sup>3</sup>) 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:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 127.7 \text{ hrs}$$

One-tenth of this time constant is approximately 12.8 hours (766 minutes), substantially longer than the fire duration of 3.622 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The temperature of the MPC is therefore increased by the contained decay heat only.

Table 4.5.5 lists lower-bound thermal inertia values for the MPC and the contained fuel assemblies of 4680 Btu/°F and 2240 Btu/°F, respectively. Applying an upper-bound decay heat load of 28.74 kW (98,090 Btu/hr) for the 3.622 minute (0.0604 hours) fire duration results in the contained fuel assemblies heating up by only:

$$\Delta T_{fuel} = \frac{98090 \times 0.0604}{4680 + 2240} = 0.86^\circ F$$

This is a negligible increase in the fuel temperature. Consequently, the impact on the MPC internal helium pressure will be negligible as well. 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 significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM 100 System to cool the spent nuclear fuel within design temperature limits during post-fire temperature relaxation is not compromised.

### Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

### Thermal

As discussed above, the MPC internal pressure increases a negligible amount and is bounded by the 100% fuel rod rupture accident in Section 11.2.9. As shown in Table 11.2.2, the peak fuel cladding



and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.

### Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: “the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR.” Less than one-inch of the concrete (less than 4% of the total overpack radial concrete section) exceeds the short-term temperature limit.

### 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 a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.4.2.2 Fire Analysis for HI-TRAC Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 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. All of these parameters are conservatively bounded by the values in Table 11.2.3, which are used for the fire transient analysis.

Using the values stated in Table 11.2.3, a bounding cask temperature rise of 5.509°F per minute is determined 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, in accordance with a Technical Specification in Appendix A to the CoC. The duration of the

50-gallon fire is 4.775 minutes. Therefore, *the temperature rise computed as the product of the rate of temperature rise and the fire duration is 26.3 °F, and the fuel cladding temperature limit is not exceeded (see Table 11.2.5).* ~~will not exceed the short-term fuel cladding temperature limit (see Table 11.2.5).~~

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. Table 11.2.4 provides the MPC maximum internal pressures, as a result of the HI-TRAC fire accident, *for a conservatively bounding initial steady state condition of the highest normal operating pressure and minimum cavity average temperature. The computed accident pressure is substantially below the accident design pressure (Table 2.2.1).* The values presented in Table 11.2.4 are determined using a bounding temperature rise of 43.2°F, instead of the calculated 26.3°F temperature rise, and are therefore conservative. Table 11.2.5 provides a summary of the loaded HI-TRAC bounding maximum temperatures for the hypothetical fire accident condition.

### Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

### Thermal

As discussed above, the MPC internal pressure *and fuel temperature* increases as a result of the fire accident. ~~but the~~ *The fire accident MPC internal pressure and peak fuel cladding temperature, conservatively including a non-mechanistic 100% fuel rod rupture, is shown in Table 11.2.4 to be are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).* ~~accident condition MPC internal design pressure of 200 psig (Table 2.2.1). As shown in Table 11.2.5, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.~~

The loss of the water in the water jacket causes the temperatures to increase ~~slightly~~ due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8 based on an assumed start at normal on-site transport conditions *and assuming that a steady state is reached.* As can be seen from the values in the table, *the temperatures are below the accident temperature limits.* ~~the temperatures increase by less than 20°F. Therefore, if the temperatures presented in Table 11.2.5 were increased by 20°F to account for the decrease in~~

conductivity of the water jacket, the resultant temperatures will still be well below the short term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

### Shielding

The assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The assumed loss of all the Holtite in the 125-ton HI-TRAC lids results in an increase in the radiation dose rates at locations adjacent to the lids. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event, since the internal pressure does not exceed the accident condition design pressure and the MPC confinement boundary temperatures do not exceed the short-term allowable temperature limits.

### Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. HI-TRAC dose rates at 1 meter and 100 meters from the water jacket, after the water is lost, have already been reported in Subsection 11.2.1.32. Immediately after the fire accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

#### 11.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The elevated temperatures experienced by the HI-STORM overpack concrete shield is limited to the outermost layer. Therefore, any corresponding reduction in neutron shielding capabilities is limited to the outermost layer. The slight increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate, due to the limited amount of the concrete shielding with reduced effectiveness and the negligible neutron dose rate calculated for normal conditions at the site boundary. The loaded HI-STORM overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC confinement boundary is not compromised and therefore, there is no release of airborne radioactive materials.

#### 11.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC or HI-STORM overpack, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC as the pressure relief valves may have opened and water loss from the water jacket may have occurred resulting in an increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, install temporary shielding around the HI-TRAC. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC. If damage to the HI-TRAC is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced by adding water at pressure. If damage to the HI-TRAC water jacket or HI-TRAC body is widespread and/or radiological conditions require, the HI-TRAC shall be unloaded in accordance with Chapter 8, prior to repair.

If damage to the HI-STORM storage overpack as the result of a fire event is widespread and/or as radiological conditions require, the MPC shall be removed from the HI-STORM overpack in accordance with Chapter 8. However, the thermal analysis described herein demonstrates that only the outermost layer of the radial concrete exceeds its design temperature. The HI-STORM overpack may be returned to service if there is no increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the visual inspection is satisfactory.

#### 11.2.5 Partial Blockage of MPC Basket Vent Holes

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. The partial blockage of the MPC basket vent holes analyzes the effects on the HI-STORM 100 System due to the restriction of the vent openings.

##### 11.2.5.1 Cause of Partial Blockage of MPC Basket Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, vacuum dried, and backfilled with helium. There are only two possible sources of material that could block the MPC basket vent holes. These are the fuel cladding/fuel pellets and crud. Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture, and that fuel cladding and fuel pellets would fall to block the basket vent holes. It is conceivable that a percentage of the crud deposited on the fuel rods may fall off of the fuel assembly and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. The HI-STORM 100 System maintains the peak fuel cladding temperature below the required long-term storage limits.

All credible accidents do not cause the fuel assembly to experience an inertia loading greater than 60g's. Therefore, there is no mechanism for the extensive rupture of spent fuel rod cladding.

Crud can be made up of two types of layers, loosely adherent and tightly adherent. The SNF assembly movement from the fuel racks to the MPC may cause a portion of the loosely adherent crud to fall away. The tightly adherent crud is not removed during ordinary fuel handling operations. The MPC vent holes that act as the bottom plenum for the MPC internal thermosiphon are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (i.e., the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. Typically, BWR plants have more crud than PWR plants. Based on the maximum expected crud volume per fuel assembly provided in reference [11.2.5], and the area at the base of the MPC basket fuel storage cell, the maximum depth of crud at the bottom of the MPC-68 was determined. For the PWR-style MPC designs (see Table 1.2.1), 90% of the maximum crud volume was used to determine the crud depth. The maximum crud depths calculated for each of the MPCs is listed in Table 2.2.8. The maximum amount of crud was assumed to be present on all fuel assemblies within the MPC. Both the tightly and loosely adherent crud was conservatively assumed to fall off of the fuel assembly. As can be seen by the values listed in the table, the maximum amount of crud depth does not totally block any of the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the available vent holes area is greater than that used in the thermal models.

#### 11.2.5.2 Partial Blockage of MPC Basket Vent Hole Analysis

The partial blockage of the MPC basket vent holes has no affect on the structural, confinement and thermal analysis of the MPC. There is no affect on the shielding analysis other than a slight increase of the gamma radiation dose rate at the base of the MPC due to the accumulation of crud. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible, and, therefore, the criticality analyses are not affected.

##### Structural

There are no structural consequences as a result of this event.

##### Thermal

There is no effect on the thermal performance of the system as a result of this event.

##### Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

## Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this accident 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 accident event.

Based on this evaluation, it is concluded that the partial blockage of MPC vent holes does not affect the safe operation of the HI-STORM 100 System.

### 11.2.5.3 Partial Blockage of MPC Basket Vent Holes Dose Calculations

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary. Therefore, there will be no effect on the site boundary dose rates because the magnitude of the radiation source has not changed. There will be no radioactive material release.

### 11.2.5.4 Partial Blockage of MPC Basket Vent Holes Corrective Action

There are no consequences that exceed normal storage conditions. No corrective action is required for the partial blockage of the MPC basket vent holes.

## 11.2.6 Tornado

### 11.2.6.1 Cause of Tornado

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. Additionally, the transfer of the MPC from the HI-TRAC transfer cask to the overpack may be performed at the unsheltered ISFSI concrete pad. It is possible that the HI-STORM System (storage overpack and HI-TRAC transfer cask) may experience the extreme environmental conditions of a tornado.

### 11.2.6.2 Tornado Analysis

The tornado accident has two effects on the HI-STORM 100 System. The tornado winds and/or tornado missile attempt to tip-over the loaded overpack or HI-TRAC transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds which attempt to penetrate the storage overpack or HI-TRAC transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3.1. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC while being handled in the vertical orientation. During handling of the loaded HI-TRAC in the horizontal orientation, it is possible that the tornado missile and/or wind may cause the rollover of the loaded HI-TRAC on the transport vehicle. The horizontal drop handling accident for the loaded HI-TRAC, Subsection 11.2.1, evaluates the consequences of the loaded HI-TRAC falling from the horizontal handling height limit and consequently this bounds the effect of the roll-over of the loaded HI-TRAC on the transport vehicle.

### Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the storage overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or tornado missiles.

Analyses provided in Section 3.4 also shows that the tornado missiles do not penetrate the storage overpack or HI-TRAC transfer cask to impact the MPC. The result of the tornado missile impact on the storage overpack or HI-TRAC transfer cask is limited to damage of the shielding.

### Thermal

The loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

### Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

## Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. There are increases in the local dose rates adjacent water jacket as a result of the loss of water in the HI-TRAC water jacket. HI-TRAC dose rates at 1 meter and 100 meters from the water jacket, after the water is lost, have already been ~~discussed~~<sup>reported</sup> in Subsection 11.2.1.32. Immediately after the tornado accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

### 11.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded storage overpack; damage the shielding materials of the overpack or HI-TRAC; or damage the MPC confinement boundary. There is no affect on the radiation dose as a result of the tornado winds. A tornado missile may cause localized damage in the concrete radial shielding of the storage overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

### 11.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM 100 System to a tornado, the ISFSI operator shall perform a visual and radiological inspection of the overpack and/or HI-TRAC transfer cask. Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and repaired. Damage sustained by the HI-TRAC shall be inspected and repaired.

## 11.2.7 Flood

### 11.2.7.1 Cause of Flood

The HI-STORM 100 System will be located on an unsheltered ISFSI concrete pad. Therefore, it is possible for the storage area to be flooded. The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

### 11.2.7.2 Flood Analysis

The flood accident affects the HI-STORM 100 overpack structural analysis in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood affects the MPC by applying an external pressure.



## Structural

Section 3.4 provides the analysis of the flood water applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

## Thermal

For a flood of sufficient magnitude to allow the water to come into contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a large flood will enter the annulus between the MPC and the overpack. The water would actually provide cooling that exceeds that available in the air filled annulus, due to water's higher thermal conductivity, density and heat capacity, and the forced convection coefficient associated with flowing water. Since the flood water temperature will be within the off-normal temperature range specified in Table 2.2.2, the thermal transient associated with the initial contact of the floodwater will be bounded by the off-normal operation conditions.

For a smaller flood that blocks the air inlet ducts but is not sufficient to allow water to come into contact with the MPC, a thermal analysis is included in Subsection 11.2.13 of this FSAR.

## Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water acts as a radiation shield and will reduce the radiation doses.

## Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the fuel pool which is presented in Section 6.1.

## Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100 System.

### 11.2.7.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

### 11.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM 100 System sustains no damage as a result of the flood. At the completion of the flood, *surfaces wetted by floodwater shall be cleared of debris and cleaned of adherent foreign matter.* ~~the exterior and interior of the overpack, and the exterior of the MPC shall be cleaned to maintain the proper air flow and emissivity.~~

## 11.2.8 Earthquake

### 11.2.8.1 Cause of Earthquake

The HI-STORM 100 System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM 100 System, the ISFSI may experience an earthquake.

### 11.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM 100 System. The objective is to determine the stability limits of the HI-STORM 100 System. Based on a static stability criteria, it is shown in Chapter 3 that the HI-STORM 100 System is qualified to seismic activity less than or equal to the values specified in Table 2.2.8. The analyses in Chapter 3 show that the HI-STORM 100 System will not tip over under the conditions evaluated. The seismic activity has no adverse thermal, criticality, confinement, or shielding consequences.

Some ISFSI sites will have earthquakes that exceed the seismic activity specified in Table 2.2.8. For these high-seismic sites, anchored HI-STORM designs (the HI-STORM 100A and 100SA) have been developed. The design of these anchored systems is such that seismic loads cannot result in tip-over or lateral displacement. Chapter 3 provides a detailed discussion of the anchored systems design.

### Structural

The sole structural effect of the earthquake is an inertial loading of less than 1g. This loading is bounded by the tip-over analysis presented in Section 11.2.3, which analyzes a deceleration of 45g's and demonstrates that the MPC allowable stress criteria are met.

### Thermal

There is no effect on the thermal performance of the system as a result of this event.

### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. If the overpack were to tip over, the resultant damage would be equal to that experienced by the tip-over accident analyzed in Subsection 11.2.3. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates or release of radioactivity.

#### 11.2.8.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have tipped-over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 11.2.3.4.

## 11.2.9 100% Fuel Rod Rupture

This accident event postulates that all the fuel rods rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity.

### 11.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. There is no credible cause for 100% fuel rod rupture. This accident is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

### 11.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no thermal, structural, criticality or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source which is being shielded, the shielding capability, or the criticality control features of the HI-STORM 100 System. The determination of the maximum accident pressure is provided in Chapter 4. The MPC design basis internal pressure bounds the pressure developed assuming 100% fuel rod rupture. The structural analysis provided in Chapter 3 evaluates the MPC confinement boundary under the accident condition internal pressure.

#### Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

#### Thermal

The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.14. As can be seen from the values, the ~~200 psig~~ design basis accident condition MPC internal pressure (*Table 2.2.1*) used in the structural evaluation bounds the calculated value.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this event.

## Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

## Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100 System.

### 11.2.9.3 100% Fuel Rod Rupture Dose Calculations

The MPC confinement boundary maintains its integrity. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. Therefore, there is no release of radioactive material or significant increase in radiation dose rates.

### 11.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC confinement boundary is not damaged. The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

### 11.2.10 Confinement Boundary Leakage

*The MPC uses redundant confinement closures to assure that there is no release of radioactive materials for postulated storage accident conditions. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The discussion contained in Chapter 7 demonstrates that MPC is designed, welded, tested and inspected to meet the guidance of ISG-18 such that leakage from the confinement boundary is considered non-credible. ~~The confinement boundary leakage accident assumes simultaneous rupture of 100% of the fuel rods and the release of the available radioactive gas inventory to the environment at a rate based on 150% of the maximum leak rate under reference conditions.~~*

#### 11.2.10.1 Cause of Confinement Boundary Leakage

There is no credible cause for confinement boundary leakage. The accidents analyzed in this chapter show that the MPC confinement boundary withstands all credible accidents. There are no man-made

or natural phenomena that could cause failure of the confinement boundary restricting radioactive material release. *Additionally, because the MPC satisfies the criteria specified in Interim Staff Guidance (ISG) 18, there is no credible leakage that would occur from the confinement boundary. The release is analyzed to demonstrate the safety of the HI-STORM 100 System*

#### 11.2.10.2 — Confinement Boundary Leakage Analysis

The following is the basis for the conservative analysis of the confinement boundary leakage accident.

1. All the fuel stored in the MPC has been cooled for 5 years. The PWR fuel type is the B&W 15×15 at 4.8% 5.0% enrichment with a burnup of 70,000 75,000 MWD/MTU. The BWR fuel type is the GE 7×7 at 4.84% enrichment with a burnup of 60,000 70,000 MWD/MTU. These fuel characteristics bound the design basis fuel for the HI-STORM 100 System.
2. One hundred percent of all the fuel rods are assumed to rupture.
3. The releasable source term and release fractions are in accordance with NUREG-1536, ISG-5 and ISG-11.

The maximum possible leakage rate of radionuclides to the environment is based on the helium leak rate under reference test conditions from the Technical Specification in Appendix A to the CoC.

*Credit is taken for the gravitational settling of fines, volatiles and crud.*

Chapter 7 presents an evaluation of the consequences of a non-mechanistic postulated ground-level breach of the MPC confinement boundary under hypothetical accident conditions of storage. The resulting Total Effective Dose Equivalent (TEDE) and other dose equivalents at a downstream distance of 100 meters are evaluated for each MPC type.

#### Structural

There are no structural consequences of the loss of confinement accident.

#### Thermal

Since this event is a non-mechanistic assumption, there are no realistic thermal consequences. As discussed in the Technical Specifications in Appendix A to the CoC, the leak test rate would result in a negligible loss of helium fill gas over the design life of the MPC, which would have an inconsequential effect on thermal performance.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

This event is based upon an assumed instantaneous breach of the confinement.

### Radiation Protection

The postulated release will result in an increase in dose to the public. The analysis of this event is provided in Section 7.3. As shown therein, the postulated breach results in dose rates to the public less than the limit established by 10CFR72.106(b) for the site boundary.

#### 11.2.10.3 Confinement Boundary Leakage Dose Calculations

10CFR72.106 requires that any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 Rem to the whole body or any organ from any design basis accident. The maximum whole body dose contribution as a result of the instantaneous leak accident is calculated in Chapter 7 (Table 7.3.8). The maximum doses as a result of the confinement boundary leak accident is calculated in Chapter 7 (Table 7.3.8). Both values are well below the regulatory limit of 5 Rem.

#### 11.2.10.34 Confinement Boundary Leakage Accident Corrective Action

*The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel. No corrective actions are required.*

A detected breached MPC will need to be repaired or the fuel removed and placed into a new MPC. First, the breached MPC must be returned to the facility in accordance with the procedures provided in Chapter 8. If the leak can be detected and repaired, and testing can be performed to verify the integrity of the confinement boundary, the MPC may be placed back into service. Otherwise, the MPC should be unloaded in accordance with the procedures provided in Chapter 8.

#### 11.2.11 Explosion

##### 11.2.11.1 Cause of Explosion

An explosion within the bounds of an ISFSI is improbable since there are no explosive materials within the site boundary. An explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. The fuel available for the explosion would be limited and therefore, any

explosion would be limited in size. Any explosion stipulated to occur beyond the site boundary would have a minimal effect on the HI-STORM 100 System.

#### 11.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external pressure of 60 psig (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 11.2.7 and the tornado missile accident of Subsection 11.2.6, because explosive materials will not be stored within close proximity to the casks. The HI-STORM Overpack does not experience the 60 psi external pressure since it is not a sealed vessel. However, a pressure differential of 10.0 psi (Table 2.2.1) is applied to the overpack. Section 3.4 provides the analysis of the accident external pressure on the MPC and overpack. The analysis shows that the MPC can withstand the effects of the accident condition external pressure, while conservatively neglecting the MPC internal pressure.

#### Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

#### Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### Shielding

There is no effect on the shielding performance of the system as a result of this event.

#### Criticality

There is no effect on the criticality control features of the system as a result of this event.

#### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

#### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100 System.



### 11.2.11.3 Explosion Dose Calculations

The bounding external pressure load has no effect on the HI-STORM 100 overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM 100 System is experienced as a result of the explosion pressure load. The effects of explosion generated missiles on the HI-STORM 100 System structure is bounded by the analysis of tornado generated missiles.

### 11.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by the explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM 100 System. Following an explosion, the ISFSI operator shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the concrete material may be replaced and the outer shell repaired.

### 11.2.12 Lightning

#### 11.2.12.1 Cause of Lightning

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad. There is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

#### 11.2.12.2 Lightning Analysis

The HI-STORM 100 System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM 100 System is hit with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, will provide a direct path to ground.

The MPC provides the confinement boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

The lightning accident shall have no adverse consequences on thermal, criticality, confinement, shielding, or structural performance of the HI-STORM 100 System.

### Structural

There is no structural consequence as a result of this event.

### Thermal

There is no effect on the thermal performance of the system as a result of this event.

### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.12.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the confinement boundary or shielding materials. Therefore, no further analysis is necessary.

#### 11.2.12.4 Lightning Accident Corrective Action

The HI-STORM 100 System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

#### 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 (in excess of 170,000 lbs), it is expected that a significant temperature rise is only possible if the completely 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. Thus, the worst possible scenario is a complete loss of ventilation air during the scheduled surveillance time interval in effect at the ISFSI site.

It is noted that there is a large thermal margin, between the maximum calculated fuel cladding temperature with design-basis fuel decay heat (Tables 4.4.9, 4.4.10, 4.4.26 and 4.4.27) and the short-term fuel cladding temperature limit (*Table 2.2.34* 58°F), to meet the transient short-term fuel cladding temperature excursion. In other words, the fuel stored in a HI-STORM system can heat up by over 300°F before the short-term peak temperature limit is reached. The concrete in the overpack and the MPC and overpack structural members also have significant margins between their calculated maximum long-term temperatures and their short-term temperature limits, with which to withstand such extreme hypothetical events.

To rigorously evaluate the minimum time available before the short-term temperature limits of either the concrete, structural members or fuel cladding are exceeded, a transient thermal model of the HI-STORM System is developed. The HI-STORM system transient model with all four air inlet ducts completely blocked is created as an axisymmetric finite-volume (FLUENT) model. With the exceptions of the inlet air duct blockage and the specification of thermal inertia properties (i.e., density and heat capacity), the model is identical to the steady-state models discussed in Chapter 4 of this FSAR. The model includes the lowest MPC thermal inertia of any MPC design.

In the first step of the transient solution, the decay heat load is set equal to 22.25 kW, and the MPC internal convection (i.e., thermosiphon) is suppressed. This evaluation provides the peak temperatures of the fuel cladding, the MPC confinement boundary and the concrete overpack shield wall, all as a function of time. Because the MPC with the lowest thermal inertia is used in the analysis, the temperature rise results obtained from evaluation of this transient model, therefore, bound the temperature rises for all MPC designs (Table 1.2.1) under this postulated event. The results of the blocked duct thermal transient evaluation are presented in ~~Figure 11.2.7 and Table 11.2.9. Figure 11.2.7 presents the temperature rise as a function of time after complete air inlet duct blockage for the following:~~

- ~~i. Fuel Cladding at the Location of Initial Maximum Temperature~~
- ~~ii. MPC Shell at the Location of Initial Maximum Temperature~~
- ~~iii. Overpack Inner Concrete at the Active Fuel Axial Mid-Height~~
- ~~iv. Overpack Inner Concrete at the Location of Initial Maximum Temperature~~
- ~~v. Overpack Outer Concrete at the Active Fuel Axial Mid-Height~~
- ~~vi. Overpack Outer Concrete at the Location of Initial Maximum Temperature~~

The concrete section average (i.e., through thickness) temperature remains below the short-term temperature limit through 72 hours of blockage. Both the fuel cladding and the MPC confinement boundary temperatures remain below their respective short-term temperature limits at 72 hours, the fuel cladding by over 150°F and the confinement boundary by almost 175°F. Table 11.2.9 summarizes the temperatures at several points in the HI-STORM System at 33 hours and 72 hours after complete inlet air duct blockage. These results establish the design-basis minimum surveillance interval (~~i.e., 24 hours per Technical Specifications in Appendix A to the CoC~~) for the duct screens. *As soon as one or more ducts are part open convection flow is re-started, convective heat dissipation begins and temperatures trend downwards to approach normal conditions as the ducts are fully cleared.*

Incorporation of the MPC thermosiphon internal natural convection, as described in Chapter 4, enables the maximum design basis decay heat load to rise to about 29 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak MPC closure plate outer surface temperature, for example, is computed to be about 450°F in the thermosiphon-enabled solution compared to about 210°F in the thermosiphon-suppressed solution, with both solutions computing approximately the same peak clad temperature. In the 100% inlet duct blockage condition, the heated MPC closure plate and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and by virtue of the fact that thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limit for reaching the short-term peak fuel cladding temperature limit (72 hours) remains applicable.

It should be noted that the rupture of 100% of the fuel rods and the subsequent release of the contained rod gases has a significant positive impact on the MPC internal thermosiphon heat transport mechanism. The increase in the MPC internal pressure accelerates the thermosiphon, as does the introduction of higher molecular weight gaseous fission products. The values reported in Table 11.2.9 do not reflect this improved heat transfer and will actually be lower than reported. Crediting the increased MPC internal pressure only and neglecting the higher molecular weights of the gaseous fission products, the MPC bulk average gas temperature will be reduced by approximately 34.5°C (62.1°F).

Under the complete air inlet ducts blockage accident condition, it must be demonstrated that the MPC internal pressure does not exceed its design-basis accident limit during this event. Chapter 4 presented the MPC internal pressure calculated at an ambient temperature of 80°F, 100% fuel rods ruptured, full insolation, and maximum decay heat. This calculated pressure is 174.8 psia, as reported in Table 4.4.14, at an average temperature of 513.6°K. Using this pressure, an ~~bounding~~

increase in the MPC cavity *bulk* temperature of 184°F (102.2°K, maximum of MPC shell or fuel cladding temperature rise 33 hours after blockage of all four ducts, see Table 11.2.9), the reduction in the bulk average gas temperature of 34.5°C, and the ideal gas law, the resultant MPC internal pressure is calculated below.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(174.8 \text{ psia})(513.6^\circ \text{K} + 102.2^\circ \text{K})}{513.6^\circ \text{K}}$$

$$P_2 = 197.8 \text{ psia or } 183.1 \text{ psig}$$

The accident MPC internal design pressure of 200 psig (Table 2.2.1) bounds the resultant pressure calculated above. Therefore, no additional analysis is required.

### Structural

There are no structural consequences as a result of this event.

### Thermal

Thermal analysis is performed to determine the time until the concrete section average and peak fuel cladding temperatures approach their short-term temperature limits. At the specified time limit, both the concrete section average and peak fuel cladding temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is calculated as presented above. As can be seen from the value above, the 200 psig design basis internal pressure for accident conditions used in the structural evaluation bounds the calculated value above.

To demonstrate the robustness of the HI-STORM System design, the results of the parametric study of incremental duct blockage performed in Subsection 11.1.4 are examined again. Even with three air inlet ducts completely blocked, as shown in Table 11.1.2, large steady-state margins against the short-term temperature limits exist for all system components and the fuel cladding of the stored assemblies. Both the peak fuel cladding and overpack concrete section average temperatures, which approach their limiting temperatures under the 100% blockage condition, with a single open duct are approximately 240°F and 100°F, respectively, less than their respective short-term temperature limits. These results show that only a relatively small amount of the total air inlet duct area, on the order of 25% or less, must remain open to prevent exceeding system short-term temperature limits under steady-state conditions.

### 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. ~~The Technical Specifications in Appendix A to the CoC specify the time interval to ensure that the blockage duration cannot exceed the time limit calculated herein.~~

#### 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 ducts accident shows that the overpack concrete section average and fuel cladding peak temperatures ~~remain substantially below their short-term temperature~~ *are within the accident temperature* limits if the blockage is cleared within 72 hours. 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. ~~Per the Technical Specifications in Appendix A to the CoC, visual inspection of the duct screens is specified on a frequency of 24 hours, or air outlet temperature monitoring is required. Therefore, an undetected blockage event could not exceed 24 hours.~~

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. A measured temperature difference between the ambient air and the exit air that exceeds the design-basis maximum air temperature rise, calculated in Section 4.4.2, will indicate blockage of the overpack air ducts.

For an accident event that completely blocks the inlet or outlet air ducts, a site-specific evaluation or analysis may be performed to demonstrate that adequate heat removal is available for the duration of the event. Adequate heat removal is defined as overpack concrete section average and fuel cladding temperatures remaining below their short term temperature limits. For those events where an evaluation or analysis is not performed or is not successful in showing that ~~fuel cladding~~ temperatures remain below *their* short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet ducts and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet ducts using pumps or fire-hoses or blowing air into the air outlet ducts using fans, to directly cool the MPC. Another example of supplemental cooling, for sufficiently low decay heat loads, would be to remove the overpack lid to increase free-surface natural convection.

#### 11.2.14 Burial Under Debris

##### 11.2.14.1 Cause of Burial Under Debris

Burial of the HI-STORM System under debris is not a credible accident. During storage at the ISFSI, there are no structures over the casks. The minimum regulatory distance of 100 meters from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM overpack air inlet ducts has already been considered in Subsection 11.2.13.

##### 11.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM System does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure encountered during the flood bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STORM System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limit during a burial under debris accident.

To demonstrate the inherent safety of the HI-STORM 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 System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short term design fuel cladding temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, and the spent nuclear fuel decay heat generation.

As stated in Subsection 11.2.13.2, there is a margin of over 300°F between the maximum calculated fuel cladding temperature and the short-term fuel cladding temperature limit. If a highly conservative 150°F is postulated as the permissible fuel cladding temperature rise for the burial under debris scenario, then a curve representing the relationship between the time required and decay heat load can be constructed. This curve is shown in Figure 11.2.6. In this figure, plots of the burial period at different levels of heat generation in the MPC are shown based on a 150°F rise in fuel cladding temperature resulting from transient heating of the HI-STORM System. Using the values stated in Table 11.2.6, the allowable time before the cladding temperatures meet the short-term fuel cladding temperature limit can be determined using:

$$\Delta t = \frac{m \times c_p \times \Delta T}{Q}$$

where:

$\Delta t$  = Allowable Burial Time (hrs)

$m$  = Mass of HI-STORM System (lb)

$c_p$  = Specific Heat Capacity (Btu/lb×°F)

$\Delta T$  = Permissible Fuel Cladding Temperature Rise (150°F)

$Q$  = Total Decay Heat Load (Btu/hr)

The allowable burial time as a function of total decay heat load ( $Q$ ) is presented in Figure 11.2.6.

The MPC cavity internal pressure under this accident scenario is bounded by the calculated internal pressure for the hypothetical 100% air inlets blockage previously evaluated in Subsection 11.2.13.2.

### Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

### Thermal

With the cladding temperature rise limited to 150°F, the corresponding pressure rise, bounded by the calculations in Subsection 11.2.13.2, demonstrates large margins of safety for the MPC vessel structural integrity. Consequently, cladding integrity and confinement function of the MPC are not compromised.



### Shielding

There is no effect on the shielding performance of the system as a result of this event.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100 System, if the debris is removed within the specified time (Figure 11.2.6). The 24-hour minimum duct inspection interval specified in the Technical Specification in Appendix A to the CoC ensures that a burial under debris condition will be detected long before the allowable burial time is reached.

#### 11.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

#### 11.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures will not exceed the short term limit if the debris is removed within 45 hours. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be

performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the performance of this corrective action.

#### 11.2.15 Extreme Environmental Temperature

##### 11.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

##### 11.2.15.2 Extreme Environmental Temperature Analysis

The accident condition considering an environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 125°F environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Section 4.4. These temperatures are conservatively calculated at an environmental temperature of 80°F. The extreme environmental temperature is 125°F, which is an increase of 45°F. Conservatively bounding temperatures for all the MPC designs are obtained and reported in Table 11.2.7. As illustrated by the table, all the temperatures are well below the accident condition design basis temperatures. The extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STORM 100 System extreme environmental temperatures meet the design requirements.

Additionally, the extreme environmental temperature generates a pressure that is bounded by the pressure calculated for the complete inlet duct blockage condition because the duct blockage condition temperatures are much higher than the temperatures that result from the extreme environmental temperature. As shown in Subsection 11.2.13.2, the accident condition pressures are below the accident limit specified in Table 2.2.1.

#### Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

### Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in Table 11.2.7. As can be seen from this table, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

### Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

### Criticality

There is no effect on the criticality control features of the system as a result of this event.

### Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

### Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100 System.

#### 11.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature and the dose calculations are equivalent to the normal condition dose rates.

#### 11.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

## 11.2.16 Supplemental Cooling System (SCS) Failure

*The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation. Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated herein.*

### 11.2.16.1 Cause of SCS Failure

*Possible causes of SCS failure are: (a) Simultaneous loss of external and backup power, or (b) Complete loss of annulus water from an uncontrolled leak or line break.*

### 11.2.16.2 Analysis of Effects and Consequences of SCS Failure

#### Structural

*See discussion under thermal evaluation below.*

#### Thermal

*In the event of a SCS failure due to (a), the following sequence of events occur:*

- i) The annulus water temperature rises to reach it's boiling temperature (~212°F).*
- ii) A progressive reduction of water level and dryout of the annulus.*

*In the event of an SCS failure due to (b), a rapid water loss occurs and annulus is replaced with air. For the condition of a vertically oriented HI-TRAC with air in the annulus, the maximum steady-state temperatures are below the accident temperature limit (1058°F) (see Subsection 11.1.6 and Table 11.1.3). For a horizontally oriented HI-TRAC with air in the annulus, the maximum steady-state temperatures are also below the accident temperature limit (see Subsection 4.5.2.1). In Supplemental Cooling LCO 3.1.4 a time limit of 24 hours is specified to upend the HI-TRAC. This places the cask system in an analyzed condition where, as cited above, the fuel cladding temperature remains below the limit.*

*To confirm that the MPC design pressure limits (Table 2.2.1) are not exceeded, a bounding gas pressure is computed assuming fuel heatup from normal temperatures (Tables 4.4.9, 4.4.10, 4.4.26 and 4.4.27) to a clad temperature limit (1058°F). For conservatism, the MPC average gas temperature is assumed to elevate from normal conditions to 1058°F. The results, summarized in Table 11.2.10, show that the MPC pressure is below the design pressure.*

#### Shielding

*There is no adverse effect on the shielding effectiveness of the system.*

### Criticality

*There is no adverse effect on the criticality control of the system.*

### Confinement

*There is no adverse effect on the confinement function of the MPC. As discussed in the evaluations above, the structural boundary pressures are within design limits.*

### Radiation Protection

*As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.*

*Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.*

#### *11.2.16.3      SCS Failure Dose Calculations*

*The event has no radiological impact because the confinement barrier and shielding integrity are not affected.*

#### *11.2.16.4      SCS Failure Corrective Action*

*In the vertical orientation the HI-TRAC is designed to withstand an SCS failure without an adverse effect on its safety functions. For a horizontally oriented HI-TRAC, LCO 3.1.4 requires HI-TRAC upending within 24 hours.*

Table 11.2.1

INTENTIONALLY DELETED

Table 11.2.2

HI-STORM 100 OVERPACK ~~MAXIMUM~~ *BOUNDING* TEMPERATURES  
AS A RESULT OF THE HYPOTHETICAL FIRE CONDITION

<b>Material/Component</b>	<b>Initial<sup>†</sup> Condition (°F)</b>	<b>During Fire (°F)</b>	<b>Post-Fire<sup>††</sup> Cooldown (°F)</b>
Fuel Cladding	691 (MPC-24) 691 (MPC-24E) 691 (MPC-32) 740 (MPC-68)	692 (MPC-24) 692 (MPC-24E) 692 (MPC-32) 741 (MPC-68)	692 (MPC-24) 692 (MPC-24E) 692 (MPC-32) 741 (MPC-68)
MPC Fuel Basket	650 (MPC-24) 650 (MPC-24E) 660 (MPC-32) 720 (MPC-68)	651 (MPC-24) 651 (MPC-24E) 661 (MPC-32) 721 (MPC-68)	651 (MPC-24) 651 (MPC-24E) 661 (MPC-32) 721 (MPC-68)
Overpack Inner Shell	195	300	195
Overpack Radial Concrete Inner Surface	195	281	282
Overpack Radial Concrete Mid-Surface	173	173	184
Overpack Radial Concrete Outer Surface	157	529	530
Overpack Outer Shell	157	570	570

---

† Bounding 195°F uniform inner surface and 157°F uniform outer surface temperatures assumed.

†† Maximum temperature during post-fire cooldown.

Table 11.2.3

SUMMARY OF INPUTS FOR HI-TRAC FIRE ACCIDENT HEAT-UP

Minimum Weight of Loaded HI-TRAC with Pool Lid (lb)	180,436
Lower Heat Capacity of Carbon Steel (Btu/lbm·°R)	0.1
Heat Capacity UO <sub>2</sub> (Btu/lbm·°R)	0.056
Heat Capacity Lead (Btu/lbm·°R)	0.031
Maximum Decay Heat (kW)	28.74
Total Fuel Assembly Weight (lb)	40,320
Lead Weight (lb)	52,478
Water Weight (lb)	7,595



Table 11.2.4

BOUNDING HI-TRAC HYPOTHETICAL  
FIRE CONDITION PRESSURES<sup>†</sup>

Condition	Pressure (psig)			
	MPC-24	MPC-24E	MPC-32	MPC-68
Without Fuel Rod Rupture	79.8	79.8	79.8	79.8
With 100% Fuel Rod Rupture	158.9	159.3	191.1	126.6 ( <del>72.48-488</del> )

---

<sup>†</sup> The reported pressures are based on temperatures that exceed the calculated maximum temperatures and are therefore slightly conservative.

Table 11.2.5

SUMMARY OF BOUNDING MPC PEAK TEMPERATURES  
DURING A HYPOTHETICAL HI-TRAC FIRE ACCIDENT CONDITION

<b>Location</b>	<b>Initial Steady State Temperature [°F]</b>	<b>Bounding Temperature Rise [°F]</b>	<b>Hottest MPC Cross Section Peak Temperature [°F]</b>
Fuel Cladding	872	26.3	898.3
Basket Periphery	600	26.3	626.3
MPC Shell	455	26.3	481.3

Table 11.2.6

SUMMARY OF INPUTS FOR ADIABATIC CASK HEAT-UP

Minimum Weight of HI-STORM 100 System (lb) (overpack and MPC)	300,000
Lower Heat Capacity of Carbon Steel (BTU/lb/°F)	0.1
Initial Uniform Temperature of Cask (°F)	740 <sup>†</sup>
Bounding Decay Heat (kW)	28.74

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<sup>†</sup> The cask is conservatively assumed to be at a uniform temperature equal to the maximum fuel cladding temperature.

Table 11.2.7

MAXIMUM TEMPERATURES CAUSED BY EXTREME ENVIRONMENTAL TEMPERATURES<sup>†</sup> [°F]

<b>Location</b>	<b>Temperature</b>	<b>Accident Temperature Limit</b>
Fuel Cladding	736 (PWR) 785 (BWR)	1058
MPC Basket	765	950
MPC Shell	396	775
Overpack Air Exit	251	N/A
Overpack Inner Shell	244	350 (overpack concrete)
Overpack Outer Shell	190	350 (overpack concrete)

---

<sup>†</sup> Conservatively bounding temperatures reported include a hypothetical rupture of 10% of the fuel rods.

Table 11.2.8

~~MAXIMUM BOUNDING MPC~~ TEMPERATURES CAUSED BY LOSS OF WATER  
FROM THE HI-TRAC WATER JACKET [°F]

<b>Temperature Location</b>	<b>Normal</b>	<b>Calculated Without Water in Water Jacket</b>	<b>Accident Condition Design Temperature</b>
Fuel Cladding	872	888	<del>1058 short-term</del>
MPC Basket	852	868	<del>950 short-term</del>
MPC Basket Periphery	600	612	<del>950 short-term</del>
MPC Shell	455	466	<del>775 short-term</del>
HI-TRAC Inner Shell	322	342	<del>400 long-term 600 short-term</del>
HI-TRAC Water Jacket Inner Surface	314	334	<del>350 long-term</del>
HI-TRAC Enclosure Shell Outer Surface	224	222	<del>350 long-term</del>
Axial Neutron Shield <sup>†</sup>	258	261	<del>300 long-term</del>

Note: ~~Where it can be shown that the temperatures are below the normal long-term condition limits, the calculated temperatures are compared to the normal long-term temperature limits for conservatism. The corresponding short-term temperature limits are higher temperatures as presented in Table 2.2.3.~~

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<sup>†</sup> Local maximum section temperature.

Table 11.2.9

## SUMMARY OF BLOCKED AIR INLET DUCT EVALUATION RESULTS

	Max. Initial Steady-State Temp. <sup>†</sup> (°F)	Temperature Rise (°F)		Transient Temperature (°F)		Short-Term Temperature Limit (°F)
		at 33 hrs	at 72 hrs	At 33 hrs	at 72 hrs	
Fuel Cladding	740	101	160	841	900	1058
MPC Shell	351	184	250	535	601	775
Overpack Inner Shell #1 <sup>††</sup>	199	113	174	312	373	600
Overpack Inner Shell #2 <sup>†††</sup>	155	193	286	348	441	600
Overpack Outer Shell	145	14	40	159	185	600
Concrete Section Average	172	79	141	251	313	350

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<sup>†</sup> Conservatively bounding temperatures reported includes a hypothetical rupture of 10% of the fuel rods.

<sup>††</sup> Coincident with location of initial maximum.

<sup>†††</sup> Coincident with active fuel axial mid-height.

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Table 11.2.10

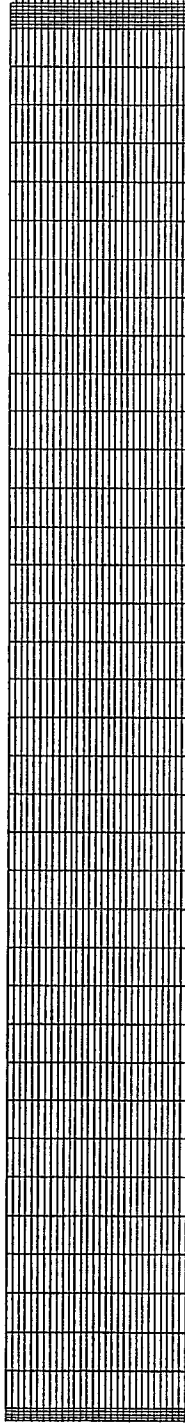
MPC PRESSURES UNDER A POSTULATED FUEL HEATUP FROM NORMAL TEMPERATURES TO ACCIDENT LIMIT (1058°F)

MPC	Normal Condition		Accident Pressure <sup>2</sup>		Design Pressure (From Chapter 2, Table 2.2.3)
	MPC Average Temperature (T <sub>o</sub> ) [°F]	Absolute Pressure (P <sub>o</sub> ) [psia] (Table 4.4.14)	Absolute (P) [psia]	Gage [psi]	
MPC-24	463	81.1	133.4	118.7	200
MPC-24E	467	80.5	131.8	117.1	200
MPC-32	464	80.3	131.9	117.2	200
MPC-68	482	81.8	131.8	117.1	200

<sup>2</sup> Conservatively assuming the MPC is heated from T<sub>o</sub> to a uniform maximum of 1058°F, the final gas pressure is computed by Ideal Gas Law as:  $P = P_o (1058 + 460)/(T_o + 460)$ .



Inner Overpack Surface Heated to 300 deg. F by Hot Gases



Outer Overpack Surface Heated by 1475 deg. F Fire Condition  
Thermal Radiation and Forced Convection

2-D Axisymmetric HI-STORM Storage Overpack Fire Transient

FIGURE 11.2.1; FIRE TRANSIENT ANSYS MODEL ELEMENT PLOT



*FIGURES 11.2.2 THROUGH 11.2.5*

*[INTENTIONALLY DELETED]*

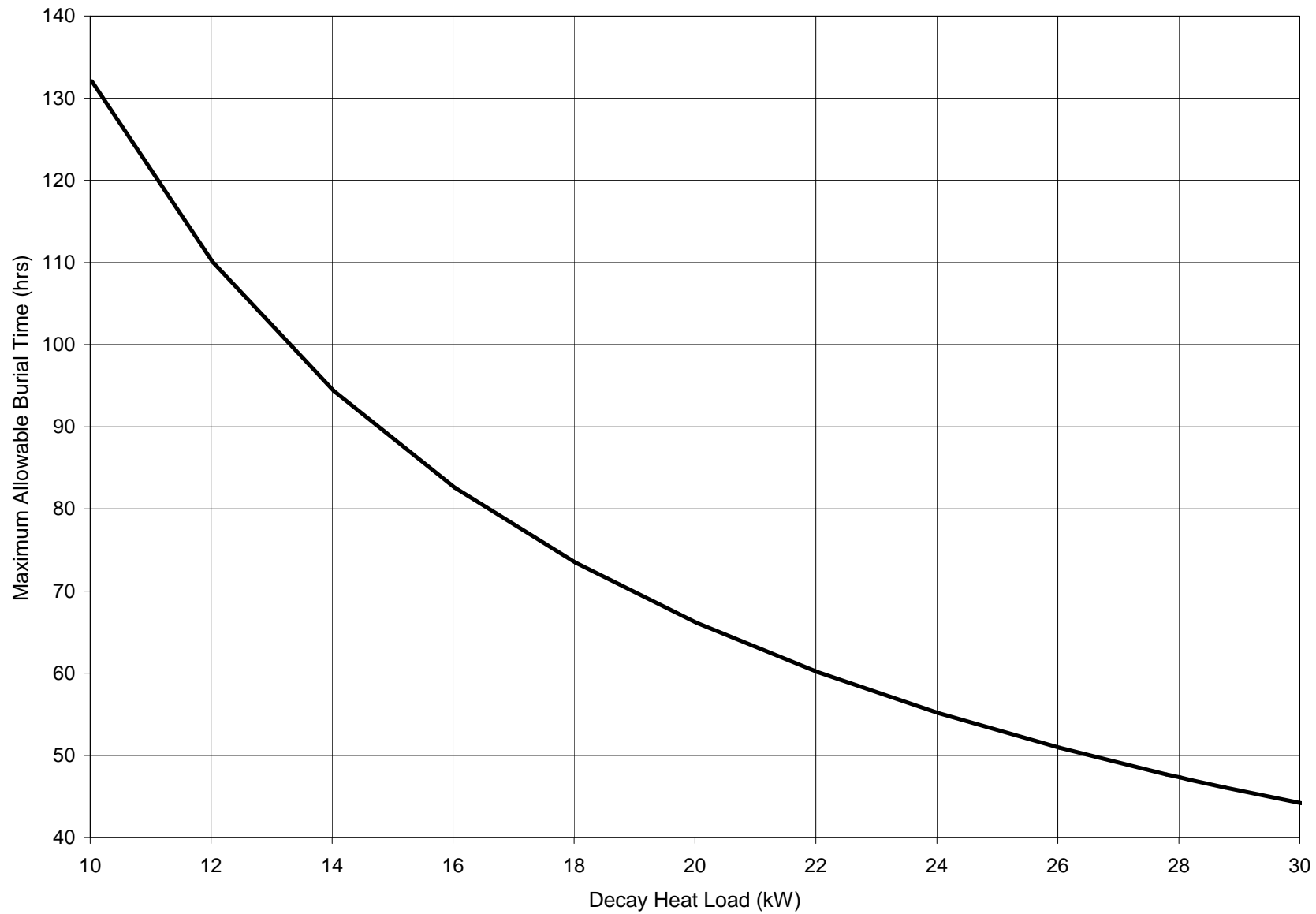


FIGURE 11.2.6; ALLOWABLE BURIAL UNDER DEBRIS TIME VERSUS DECAY HEAT LOAD

*FIGURE 11.2.7*

*[INTENTIONALLY DELETED]*

11.3        REFERENCES

- [11.2.1]     Chun, et al., “Dynamic Impact Effects on Spent Fuel Assemblies,” Lawrence Livermore National Laboratory, UCID-21246, (October 1987).
  
- [11.2.2]     Gregory, J.J., et. al., “Thermal Measurements in a Series of Large Pool Fires,” SAND85-1096, Sandia National Laboratories, Albuquerque, NM, (August 1987).
  
- [11.2.3]     IAEA Safety Standards, “Regulations for the Safe Transport of Radioactive Material,” International Atomic Energy Agency, Vienna, (1985).
  
- [11.2.4]     Deleted.
  
- [11.2.5]     ESEERCO Project EP91-29 and EPRI Project 3100-02, “Debris Collection System for Boiling Water Reactor Consolidation Equipment,” B&W Fuel Company, (October 1995).
  
- [11.2.6]     Docket Number 72-1008, HI-STAR 100 System FSAR, Holtec Report HI-2012610, latest revision.
  
- [11.2.7]     Docket Number 71-9261, HI-STAR 100 System SAR, Holtec Report HI-951251, latest revision.

## CHAPTER 12<sup>†</sup>: OPERATING CONTROLS AND LIMITS

### 12.0 INTRODUCTION

The HI-STORM 100 System provides passive dry storage of spent fuel assemblies in interchangeable MPCs with redundant multi-pass welded closure. The loaded MPC is enclosed in a single-purpose ventilated metal-concrete overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STORM 100 System at an ISFSI. The information provided in this Chapter is in full compliance with NUREG-1536 [12.1.1].

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

## 12.1 PROPOSED OPERATING CONTROLS AND LIMITS

### 12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

12.1.1.1 This portion of the FSAR establishes the commitments regarding the HI-STORM 100 System and its use. Other 10CFR72 [12.1.2] and 10CFR20 [12.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [12.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM 100 System. The general license conditions governed by 10CFR72 [12.1.2] are not repeated with these Technical Specifications. Licensees are required to comply with all commitments and requirements.

12.1.1.2 The Technical Specifications provided in Appendix A to CoC 72-1014 and the authorized contents and design features provided in Appendix B to CoC 72-1014 are primarily established to maintain subcriticality, confinement boundary and intact fuel cladding integrity, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 12.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 12.1.2 provides the list of Technical Specifications for the HI-STORM 100 System.

Table 12.1.1  
HI-STORM 100 SYSTEM CONTROLS

Condition to be Controlled	Applicable Technical Specifications <sup>†</sup>
Criticality Control	<del>Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features</del> 3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) 3.1.4 <i>Supplemental Cooling System</i> <del>5.6 Fuel Cladding Oxide Thickness Evaluation Program</del> <del>(CoC 72-1014, Appendix B Design Features)</del>
Shielding and Radiological Protection	<del>Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features</del>  3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down 3.2.1 <del>TRANSFER CASK Average Surface Dose Rates</del> 3.2.2 TRANSFER CASK Surface Contamination <del>3.2.3 OVERPACK Average Surface Dose Rates</del> 5.7 <i>Radiation Protection Program</i>
Heat Removal Capability	<del>Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features</del>  3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System 3.1.4 <i>Supplemental Cooling System</i>
Structural Integrity	3.5 Cask Transfer Facility (CTF) ( <del>CoC 72-1014, Appendix B Design Features</del> ) 5.5 Cask Transport Evaluation Program

<sup>†</sup> Technical Specifications are located in Appendix A to CoC 72-1014. *Authorized contents are specified in FSAR Section 2.1.9*

Table 12.1.2  
HI-STORM 100 SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION 1.1 Definitions 1.2 Logical Connectors 1.3 Completion Times 1.4 Frequency
2.0	Not Used. Refer to Appendix B to CoC 72-1014 for fuel specifications.
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.1.4	<i>Supplemental Cooling System</i>
3.2.1	<del>TRANSFER CASK Average Surface Dose Rates Deleted</del>
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	<del>OVERPACK Average Surface Dose Rates Deleted</del>
3.3.1	Boron Concentration
Table 3-1	MPC Model-Dependent Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used. Refer to Appendix B to CoC 72-1014 for design features.
5.0	ADMINISTRATIVE CONTROLS AND PROGRAMS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	<del>Fuel Cladding Oxide Thickness Evaluation Program Deleted</del>
5.7	<i>Radiation Protection Program</i>
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements



## 12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, *and training requirements* for the HI-STORM 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR. ~~In addition to the controls and limits provided in the Technical Specifications contained in Appendix A to Certificate of Compliance 72-1014 and the Approved Contents and Design Features in Appendix B to Certificate of Compliance 72-1014, the licensee shall ensure that the following training and dry run activities are performed.~~

### 12.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM 100 Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM 100 System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important to Safety (overview)
4. HI-STORM 100 System Final Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STORM 100 Technical Specifications, Approved Contents, Design Features and other Conditions for Use;
8. HI-STORM 100 Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Operating Experience Reviews

11. HI-STORM 100 System and ISFSI Procedures, including

- Procedural overview
- Fuel qualification and loading
- MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
- MPC welding operations
- HI-TRAC/overpack closure
- Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, *supplemental cooling*, and cooldown)
- MPC/HI-TRAC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

12.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM 100 System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The dry run shall include, but is not limited to the following:

1. Receipt inspection of HI-STORM 100 System components.
2. Moving the HI-STORM 100 MPC/HI-TRAC into the spent fuel pool.
3. Preparation of the HI-STORM 100 System for fuel loading.
4. Selection and verification of specific fuel assemblies to ensure type conformance.
5. Locating specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.
6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.
7. Replacing the HI-TRAC pool lid with the transfer lid (HI-TRAC 100 and 125 only).
8. MPC welding, NDE inspections, ~~hydrostatic pressure testing~~, draining, moisture removal, ~~and helium backfilling and leakage testing~~ (for which a mockup may be used).

9. HI-TRAC upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.
10. Placement of the HI-STORM 100 System at the ISFSI.
11. HI-STORM 100 System unloading, including cooling fuel assemblies, flooding the MPC cavity, and removing MPC welds (for which a mock-up may be used).
12. *Installation and operation of the Supplemental Cooling System.*

#### 12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STORM 100 System is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system to verify operability of the overpack heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

#### 12.2.4 Limiting Conditions for Operation

Limiting Conditions for Operation specify the minimum capability or level of performance that is required to assure that the HI-STORM 100 System can fulfill its safety functions.

#### 12.2.5 Equipment

The HI-STORM 100 System and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown in this FSAR that no credible condition or event prevents the HI-STORM 100 System from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

#### 12.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM 100 System fulfills its safety functions, provided that the Technical Specifications ~~in Appendix A to CoC 72-1014~~ and the Authorized Contents ~~and Design Features in Appendix B to CoC 72-1014~~ described in Section 2.1.9 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

#### 12.2.7 Design Features

This section describes HI-STORM 100 System design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in

this FSAR and in Appendix B to CoC 72-1014, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM 100 System is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 9.

#### 12.2.8 MPC

- a. Basket material composition, properties, dimensions, and tolerances for criticality control.
- b. Canister material mechanical properties for structural integrity of the confinement boundary.
- c. Canister and basket material thermal properties and dimensions for heat transfer control.
- d. Canister and basket material composition and dimensions for dose rate control.

#### 12.2.9 HI-STORM Overpack

- a. HI-STORM overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations.
- b. HI-STORM overpack material thermal properties and dimensions for heat transfer control.
- c. HI-STORM overpack material composition and dimensions for dose rate control.

#### 12.2.10 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

*This example executes the methodology and equations described in Section 2.1.9.1 for determining allowable decay heat, burnup, and cooling time for the approved cask contents. 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.*

*Step 1: Determine the maximum allowable assembly decay heat load for each region.*

$$q_{\text{Region 1}} = 1.131 \text{ kW}$$

$$q_{\text{Region 2}} = 0.600 \text{ kW}$$

*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. 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 using Equation 2.1.9.3.*

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.% <sup>235</sup>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 4.6 years is calculated by linear interpolation between the values in Table 12.2.1 for 4 years and 5 years of cooling. Linear interpolation yields a maximum burnup of 39,843 MWD/MTU (rounded down from 39,843.4), 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 than 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.

Table 12.2.1

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-32, Array/Class 16x16A and Enrichment = 3.1 wt.% <sup>235</sup>U)  
 ( $q_{\text{Region 1}} = 1.131 \text{ kW}$ ,  $q_{\text{Region 2}} = 0.600 \text{ kW}$ )

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)</b>
≥3	24432	12303
≥4	35110	19318
≥5	42999	24991
≥6	48530	29209
≥7	52394	32135
≥8	55322	34318
≥9	57636	36005
≥10	59584	37395
≥11	61262	38552
≥12	62786	39584
≥13	64206	40507
≥14	65551	41368
≥15	66881	42200
≥16	68184	42998
≥17	68200	43769
≥18	68200	44538
≥19	68200	45292
≥20	68200	46055

Table 12.2.2

*EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING  
 (MPC-32, Array/Class 16x16A and Enrichment = 4.185 wt.% <sup>235</sup>U)  
 ( $q_{Region 1} = 1.131$  kW,  $q_{Region 2} = 0.600$  kW)*

<b>MINIMUM COOLING TIME (years)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)</b>	<b>MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)</b>
≥3	25811	12639
≥4	36903	19962
≥5	44965	25702
≥6	50602	29910
≥7	54568	32830
≥8	57592	35020
≥9	59984	36710
≥10	62016	38132
≥11	63766	39321
≥12	65351	40372
≥13	66822	41330
≥14	68200	42224
≥15	68200	43086
≥16	68200	43913
≥17	68200	44698
≥18	68200	45497
≥19	68200	46279
≥20	68200	47067

Table 12.2.3

*SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE  
(Array/Class 16x16A)*

<b>FUEL ASSEMBLY NUMBER</b>	<b>ENRICHMENT (wt. % <sup>235</sup>U)</b>	<b>FUEL ASSEMBLY BURNUP (MWD/MT U)</b>	<b>FUEL ASSEMBLY COOLING TIME (years)</b>	<b>FUEL ASSEMBLY DECAY HEAT (kW)</b>	<b>NON-FUEL HARDWARE STORED WITH ASSEMBLY</b>	<b>NFH DECAY HEAT (kW)</b>
<i>1</i>	<i>3.0</i>	<i>37100</i>	<i>4.7</i>	<i>0.7</i>	<i>BPRA</i>	<i>0.3</i>
<i>2</i>	<i>3.2</i>	<i>38812</i>	<i>4.6</i>	<i>0.9</i>	<i>NA</i>	<i>NA</i>
<i>3</i>	<i>4.3</i>	<i>41976</i>	<i>18.2</i>	<i>0.4</i>	<i>BPRA</i>	<i>0.1</i>



### 12.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM 100 System are provided in Appendix A to Certificate of Compliance 72-1014. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to CoC 72-1014. Bases applicable to the Technical Specifications are provided in FSAR Appendix 12.A. The format and content of the HI-STORM 100 System Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, “Writer’s Guide for the Restructured Technical Specifications” [12.3.1] was used as a guide in the development of the Technical Specifications and Bases.

## 12.4 REGULATORY EVALUATION

Table 12.1.2 lists the Technical Specifications for the HI-STORM 100 System. The Technical Specifications are detailed in Appendix A to Certificate of Compliance 72-1014. The Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to CoC 72-1014.

The conditions for use of the HI-STORM 100 System identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and cask design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM 100 System will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

12.5           REFERENCES:

- [12.1.1]       U.S. Nuclear Regulatory Commission, NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, Final Report, January 1997.
- [12.1.2]       U.S. Code of Federal Regulations, Title 10, Energy, Part 72, Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."
- [12.1.3]       U.S. Code of Federal Regulations, Title 10, Energy, Part 20, Standards for Protection Against Radiation."
- [12.3.1]       Nuclear Management and Resources Council, Inc. – Writer's Guide for the Restructured Technical Specifications, NUMARC 93-03, February 1993.

**HI-STORM 100 SYSTEM FSAR**

**APPENDIX 12.A**

**TECHNICAL SPECIFICATION BASES**

**FOR THE HOLTEC HI-STORM 100 SPENT FUEL STORAGE CASK SYSTEM**

**(50 PAGES, INCLUDING THIS PAGE)**

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## B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

### BASES

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LCOs LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

---

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).

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LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the

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BASES

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LCO 3.0.2  
(continued)

remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

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LCO 3.0.3

This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

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LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM 100 System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

- a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to

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BASES

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LCO 3.0.4  
(continued)

exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continuing with dry fuel storage activities for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the dry storage system. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

---

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing to demonstrate:

- a. The equipment being returned to service meets the LCO; or
- b. Other equipment meets the applicable LCOs.

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BASES

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LCO 3.0.5  
(continued)

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

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## B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

### BASES

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SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

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SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

- a. The systems or components are known to not meet the LCO, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the HI-STORM 100 System is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances

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BASES

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SR 3.0.1  
(continued)

are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary dry storage cask system parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow dry fuel storage activities to proceed to a specified condition where other necessary post maintenance tests can be completed.

---

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension

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BASES

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SR 3.0.2  
(continued)

to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

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SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of HI-STORM 100 System conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

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BASES

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SR 3.0.3  
(continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe conduct of dry fuel storage activities.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is

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## BASES

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SR 3.0.4  
(continued)

outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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## B 3.1 SFSC Integrity

## B 3.1.1 Multi-Purpose Canister (MPC)

**BASES**

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**BACKGROUND** A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where ~~dose rates are measured and~~ the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and ~~moisture removal~~ *drying* is performed. The MPC cavity is backfilled with helium. ~~Additional dose rates are measured and~~ *Then*, the MPC vent and drain *port* cover plates and closure ring are installed and welded. Inspections are performed on the welds. ~~TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.~~

MPC cavity moisture removal using vacuum drying or forced helium ~~recirculation~~ *dehydration* is performed to remove residual moisture from the MPC ~~fuel~~ cavity *space* after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel and by the heat added to the MPC from the optional warming pad, if used.

If *forced* helium ~~dehydration~~ *recirculation* is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

(continued)

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BASES

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## BACKGROUND

(continued)

After the completion of ~~moisture removal~~ *drying*, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. ~~Providing a~~ *Backfilling the MPC with* helium ~~pressure in the required range at room temperature (70°F) quantity;~~ eliminates air leakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage. ~~Providing helium in the required density range accomplishes the same function.~~

In-leakage of air could be harmful to the fuel. Prior to moving the SFSC to the storage pad, the MPC helium leak rate is determined to ensure that the fuel is confined.

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APPLICABLE  
SAFETY  
ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC overpressurization evaluation.

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(continued)



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**BASES (continued)**

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LCO A dry, helium filled and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the multiple confinement boundaries. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.

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APPLICABILITY The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure both the confinement barriers and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored fuel.

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ACTIONS A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the cavity vacuum drying pressure or demoinsturizer exit gas temperature limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

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BASES

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ACTIONS  
(continued)

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the helium backfill ~~quantity density or pressure~~ limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

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**BASES**

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**ACTIONS**  
(continued)B.2

Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

~~C.1~~

~~If the helium leak rate limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM OVERPACK is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses calculated in the FSAR confinement analyses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.~~

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(continued)

## BASES

ACTIONS  
(continued)C.2

~~Once the cause and consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and off-site doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.~~

DC.1

If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid (*if required*), perform fuel cooldown operations (*if required*), re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1, and SR 3.1.1.2, and SR 3.1.1.3

~~SR 3.1.1.1 is modified by a note that states, in addition to the requirements of SR 3.1.1.1 for high burnup fuel, MPCs with heat loads in excess of a certain value shall be dried using the helium recirculation method. The basis for this note is that, if vacuum drying were used for higher heat load MPCs, it would need to be completed in a relatively short period of time to avoid exceeding the short term peak fuel cladding temperature limit. Applying a time limit that is too restrictive could inhibit the ability to dry the MPC in a normal time frame. The helium recirculation method of moisture removal continuously cools the fuel while removing moisture, thereby eliminating the need to establish a time limit, allowing completion of the moisture removal process in a deliberate, controlled manner.~~

## BASES

SURVEILLANCE REQUIREMENTS SR 3.1.1.1, and SR 3.1.1.2, and SR 3.1.1.3 (continued)

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. For moderate burnup fuel cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the *gas temperature or dew point at the specified location* ~~demoisturizer exit temperature~~ reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. For high burnup fuel *and high decay heat load MPCs*, the forced helium ~~recirculation~~ *dehydration* method of moisture removal must be used to provide necessary cooling of the fuel during drying operations. Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations.

*Table 3-1 of Appendix A to the CoC provides the appropriate requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high) and the applicable short-term temperature limit. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11.*

Having the proper *quantity of helium in the MPC* ~~backfill density or pressure~~ ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC *and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC.*

~~Meeting the helium leak rate limit ensures there is adequate helium in the MPC for long term storage and the leak rate assumed in the confinement analyses remains bounding for off-site dose.~~

(continued)

BASES

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SURVEILLANCE SR 3.1.1.1, and SR 3.1.1.2, and SR 3.1.1.3 (continued)  
REQUIREMENTS

~~The leakage rate acceptance limit is specified in units of atm-cc/sec. This is a mass-like leakage rate as specified in ANSI N14.5 (1997). This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions. This allows the leakage rate as measured by a mass spectrometer leak detector (MSLD) to be compared directly to the acceptance limit without the need for unit conversion from test conditions to standard, or reference conditions.~~

~~All three~~Both of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the cask design.

- 
- REFERENCES
1. FSAR Sections 1.2, 4.4, 4.5, 7.2, 7.3 and 8.1
  2. *Interim Staff Guidance Document 11*
  3. *Interim Staff Guidance Document 18*
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## B 3.1 SFSC Integrity

### B 3.1.2 SFSC Heat Removal System

#### BASES

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**BACKGROUND** The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system ~~which~~*that* ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the ~~four~~ inlet air ducts at the bottom of the OVERPACK. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the ~~four~~ outlet air ducts at the top of the OVERPACK.

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**APPLICABLE SAFETY ANALYSIS** The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the ~~four~~ inlet and ~~four~~ outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.

Analyses have been performed for the complete obstruction of ~~two~~*half*, ~~three~~, and ~~four~~*all* inlet air ducts. Blockage of ~~two~~*half of the* inlet air ducts reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits.

~~Blockage of three inlet air ducts further reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this accident condition, no SFSC components exceed the short term temperature limits.~~

(continued)

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BASES

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APPLICABLE  
SAFETY  
ANALYSIS  
(continued)

The complete blockage of all four inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler inner shell of the OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the 72-hour duration of the analyzed event. ~~Therefore, the limiting component is assumed to be the fuel cladding.~~

---

LCO

The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.

The intent of this LCO is to address those occurrences of air duct blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris).

(continued)

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BASES

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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.

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APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate ~~heat transfer~~ *dissipation* of the decay heat ~~away~~ from the fuel assemblies.

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ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

(continued)

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BASES

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ACTIONS  
(continued)

B.1

If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures . Therefore, ~~Surveillance Requirement (SR) 3.2.3.1 dose rates~~ *is* are required to be ~~performed-measured to determine-verify~~ the effectiveness of the radiation shielding provided by the concrete. This ~~SR-Action~~ *Action* must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of ~~whether~~ *the effectiveness of the* concrete ~~is providing adequate 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.

B.2.1

In addition to Required Action B.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 B.2.2 is being implemented.

This Required Action must be complete in ~~48-64~~ hours.

(continued)

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BASES

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ACTIONS

B.2.1 (continued)

The Completion Time reflects *the 8 hours to complete Required Action A.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all inlet or outlet ducts becoming simultaneously blocked by trash or debris. a conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short term temperature limit for more than 72 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.*

B.2.2

In lieu of implementing Required Action B.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.*

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

(continued)

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BASES

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ACTIONS

B.2.2 (continued)

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 Time of ~~48-64~~ hours reflects *the Completion Time from Required Action B.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads.* ~~a conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short term temperature limit for more than 72 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.~~

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.2.1

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 3.1.2.1, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all ~~four~~ inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Complete blockage of any one or more inlet or outlet air ducts renders the heat removal system inoperable and this LCO not met. Partial blockage of one or more inlet or outlet air ducts does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s).

(continued)

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BASES

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SURVEILLANCE REQUIREMENTS SR 3.1.2.1 (continued)

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limits 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.

The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.

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- REFERENCES
1. FSAR Chapter 4
  2. FSAR Sections 11.2.13 and 11.2.14
  3. ANSI/ANS 57.9-1992
-

B 3.1 SFSC INTEGRITY

B 3.1.3 Fuel Cool-Down

**BASES**

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**BACKGROUND** In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the cask preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. *The bulk helium temperature in the MPC cavity is ensured to be less than or equal to 200°F. This is accomplished via direct measurement of the MPC gas exit temperature or any other appropriate means based on a thermal evaluation of the particular MPC to be unloaded, considering its contents and the duration of time the MPC has been loaded. It is possible that the thermal evaluation may determine that the bulk gas temperature is already within the LCO limit due to low decay contents and/or an extended time since loading, in which case, no additional action is required.* ~~attached to the Cool-Down System. The Cool-Down System is a closed-loop forced-ventilation gas-cooling system that cools the fuel assemblies by cooling the surrounding helium gas.~~

*After ensuring the MPC cavity bulk helium temperature meets the LCO limit, Following fuel cool-down, the MPC is then re-flooded with water and the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and transfer cask are removed from the spent fuel pool and decontaminated.*

*Ensuring that Reducing the bulk helium temperature is less than the LCO limit fuel cladding temperatures significantly reduces the temperature gradients across the fuel cladding thus minimizing thermally-induced stresses on the cladding during MPC re-flooding. Reducing the MPC internal temperatures eliminates the risk of high MPC pressure due to sudden generation of large steam quantities during re-flooding. The LCO limit of 200°F for bulk helium temperature eliminates the potential for gross steam generation during re-flooding.*

(continued)

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BASES

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APPLICABLE  
SAFETY  
ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on minimizing thermally-induced stresses to the cladding.

This is accomplished during the unloading operations by lowering the MPC ~~internal cavity~~ *bulk helium* temperatures prior to MPC re-flooding. The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by reducing the MPC internal temperatures such that there is no sudden formation of *large quantities of* steam during MPC re-flooding. (Ref. 1).

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LCO

~~Monitoring-Determining the circulating MPC gas exit~~ *cavity bulk helium* temperature *prior to re-flooding* ensures that there will be no large thermal gradient across the fuel assembly cladding during re-flooding which could be potentially harmful to the cladding. The temperature limit specified in the LCO was selected to ensure that the MPC *cavity bulk helium temperature is sufficiently low to preclude high thermal stresses in the fuel cladding during gas exit* temperature will closely match the desired fuel cladding temperature prior to re-flooding of the MPC. The temperature was selected to be lower than the boiling temperature of water with an additional margin.

*For the purposes of this LCO, "bulk helium temperature" is defined as the spatial average of the helium temperature in the MPC cavity. The bulk helium temperature will be between the highest and lowest fuel cladding temperature present in the basket.*

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(continued)

BASES

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APPLICABILITY The MPC *cavity bulk* helium ~~gas exit~~ temperature is measured during UNLOADING OPERATIONS after the transfer cask and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the Fuel Cool-Down LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS.

A note has been added to the APPLICABILITY for LCO 3.1.3 which states that the Applicability is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concern for dry UNLOADING OPERATIONS.

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ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the MPC *cavity bulk* helium ~~gas exit~~ temperature limit is not met, actions must be taken to restore the parameters to within the limits before re-flooding the MPC. Failure to successfully complete fuel cool-down could have several causes, such as failure of the cool down system, inadequate cool down, or clogging of the piping lines. The Completion Time is sufficient to determine and correct most failure mechanisms and proceeding with activities to flood the MPC cavity with water are prohibited.

(continued)



BASES

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ACTIONS  
(continued)

A.2

If the LCO is not met, in addition to performing Required Action A.1 to restore the gas temperature to within the limit, the user must ensure that the proper conditions exist for the transfer of heat from the MPC to the surrounding environs to ensure the fuel cladding remains below the short term temperature limit. ~~If the TRANSFER CASK is located in a relatively open area such as a typical refuel floor, no additional actions are necessary.~~

~~However, if the TRANSFER CASK is located in a structure such as a decontamination pit or fuel vault, additional actions may be necessary depending on the heat load of the stored fuel.~~

~~Three acceptable options for ensuring adequate heat transfer for a TRANSFER CASK located in a pit or vault are provided below, based on an MPC loaded with fuel assemblies with design basis heat load in every storage location. Users may develop other alternatives on a site specific basis, considering actual fuel loading and decay heat generation.~~

(continued)

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BASES

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ACTIONS

A.2 (continued)

1. ~~Ensure the annulus between the MPC and the TRANSFER CASK is filled with water. This places the system in a heat removal configuration which is bounded by the FSAR thermal evaluation of the system considering a vacuum in the MPC. The system is open to the ambient environment which limits the temperature of the ultimate heat sink (the water in the annulus) and, therefore, the MPC shell to 212° F.~~
2. ~~Remove the TRANSFER CASK from the pit or vault and place it in an open area such as the refuel floor with a reasonable amount of clearance around the cask and not near a significant source of heat.~~

*Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.*

(continued)

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BASES

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ACTIONS

A.2 (continued)

~~Twenty two (22) hours is an acceptable time frame to allow for completion of Required Action A.2 based on a thermal evaluation of a TRANSFER CASK located in a pit or vault. In such a configuration, passive cooling mechanisms will be largely diminished. Eliminating 90% of the passive cooling mechanisms with the cask emplaced in the vault, the thermal inertia of the cask (approximately 20,000 Btu/° F) will limit the rate of temperature rise with design basis maximum heat load to approximately 4.5 degrees F per hour. Thus, the fuel cladding temperature rise in 22 hours will be less than 100° F. Large short term temperature margins exist to preclude any cladding integrity concerns under this temperature rise.~~

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on the material condition of the fuel assembly cladding. By minimizing thermally-induced stresses across the cladding the integrity of the fuel assembly cladding is maintained. The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal temperature prior to re-flooding the MPC there is minimal formation of steam during MPC re-flooding.

The MPC *cavity bulk* helium ~~exit gas~~ temperature limit ensures that there will be no large thermal gradients across the fuel assembly cladding during MPC re-flooding and ~~minimal~~ formation of steam which could potentially overpressurize the MPC.

(continued)

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BASES

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SURVEILLANCE REQUIREMENTS SR 3.1.3.1 (continued)

*The SR is met in one of two ways. The temperature of the gas exiting the MPC may be measured directly. Alternatively, a thermal evaluation may be performed, consistent with the methodology in the HI-STORM FSAR, to determine the MPC bulk helium temperature in the canister designated for unloading. This evaluation may consider the particular characteristics of the MPC, such as fuel cooling time, presence of NON-FUEL HARDWARE, and ambient conditions in determining the bulk helium temperature. If the MPC cavity bulk helium temperature LCO is shown to be met by this evaluation, no further actions are required and MPC unloading may proceed. If the LCO is shown not to be met by the thermal evaluation, appropriate means shall be used to cool the MPC cavity until the LCO is met (via direct measurement of the helium gas exit temperature or by an evaluation that includes the cooling process). When the LCO is met, unloading may proceed.*

*The LCO must be met ~~Fuel cool down must be performed successfully~~ on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved.*

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REFERENCES 1. FSAR, Sections 4.4.1, ~~4.5.1.1.4~~, 4.4, 4.5 and 8.3.2.

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B 3.1 SFSC Integrity

B 3.1.4 Supplemental Cooling System

**BASES**

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**BACKGROUND**     *The Supplemental Cooling System (SCS) is an active, water cooling system that provides augmented heat removal from the MPC to ensure fuel cladding temperatures remain below the applicable limit during onsite transport operations in the TRANSFER CASK. The system is required for all MPCs meeting the burnup, heat load, and TRANSFER CASK orientation combinations specified in the Applicability of the LCO.*

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**APPLICABLE SAFETY ANALYSIS**     *The thermal analyses of the MPC inside the TRANSFER CASK take credit for the operation of the SCS under certain conditions to ensure that the spent fuel cladding temperature remains below the applicable limit. FSAR Section 4.5 describes these analyses in more detail. For MPCs containing all moderate burnup fuel ( $\leq 45,000$  MWD/MTU), SCS operation is not required, because the fuel cladding temperature cannot exceed the limit of 1058°F for moderate burnup fuel (Refs. 2 and 3).*

*For high burnup fuel, the fuel cladding temperature limit is 400°C (752°F) during onsite transportation. For MPCs containing one or more high burnup fuel assemblies, the SCS has been credited in the thermal analysis in order to meet the lower fuel cladding temperature limit.*

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*(continued)*

BASES

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**LCO**                      *The Supplemental Cooling System must be operable if the MPC/TRANSFER cask assemblage meets one of the following conditions in the Applicability portion of the LCO in order to preserve the assumptions made in the thermal analysis.*

---

**APPLICABILITY**      *The LCO is applicable within 4 hours after completion of MPC drying operations in accordance with LCO 3.1.1 or within 4 hours of transferring the MPC into the TRANSFER CASK if the MPC is to be unloaded, and the following conditions are met:*

*MPCs having one or more fuel assemblies with an average burnup greater than 45,000 MWD/MTU.*

---

**ACTIONS**              A.1

*If the SCS has been determined to be inoperable, the thermal analysis shows that the fuel cladding temperature would not exceed the short term temperature limit applicable to an off-normal condition, even with no water in the TRANSFER CASK-to-MPC annulus. Actions should be taken to restore the SCS to operable status in a timely manner. Because the thermal analysis is a steady-state analysis, there is an indefinite period of time available to make repairs to the SCS. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 7 days is considered appropriate and a reasonable amount of time to plan the work, obtain needed parts, and execute the work in a controlled manner.*

(continued)

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**BASES**

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**ACTIONS**  
(continued)

B.1

*If, after 7 days, the SCS cannot be restored to operable status, actions should be taken to remove the fuel assemblies from the MPC and place them back into the spent fuel pool storage racks. Thirty days is considered a reasonable time frame given that the MPC will be adequately cooled while this action is being planned and implemented, and certain equipment for this infrequent evolution (e.g., weld cutting machine) may take some time to acquire.*

---

**SURVEILLANCE**  
**REQUIREMENTS** SR 3.1.4.1

*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, including during short-term evolutions such as on-site transportation in the TRANSFER CASK. The SCS is required to ensure adequate fuel cooling in certain cases. The SCS should be verified to be operable every two hours. This would involve verification that the water flow rate and temperatures are within expected ranges and the pump and air cooler are operating as expected. This is a reasonable Frequency given the typical oversight occurring during the on-site transportation evolution, the duration of the evolution, and the simple equipment involved.*

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**REFERENCES**

1. FSAR Section 4.5
  2. NRC Interim Staff Guidance 11, Rev. 3
  3. NRC Memorandum, C. Brown to M.W. Hodges, January 29, 2004
-

B 3.2 ~~SFSC Radiation Protection Deleted~~

B 3.2.1 ~~TRANSFER CASK Average Surface Dose Rates Deleted~~

**~~BASES~~**

---

~~BACKGROUND~~ — The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions.

---

~~APPLICABLE SAFETY ANALYSIS~~ — The TRANSFER CASK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.

---

~~LCO~~ — The limits on TRANSFER CASK average surface dose rates are based on the shielding analysis of the HI-STORM 100 System (Ref. 2). The limits were selected to minimize radiation exposure to the general public and maintain occupational dose ALARA to personnel working in the vicinity of the TRANSFER CASKs. The LCO requires specific locations for taking dose rate measurements to ensure the dose rates measured are indicative of the neutron shielding material's effectiveness and not the steel channel members.

---

~~APPLICABILITY~~ — The average TRANSFER CASK surface dose rates apply during ~~TRANSPORT OPERATIONS, AND UNLOADING OPERATIONS~~. These limits ensure that the transfer cask average surface dose rates during ~~TRANSPORT OPERATIONS, AND UNLOADING OPERATIONS~~ are within the estimates contained in the HI-STORM 100 Topical Safety Analysis Report. Radiation doses during ~~STORAGE OPERATIONS~~ are verified for the OVERPACK under LCO 3.2.3 and monitored thereafter by the SFSC user in accordance with the plant specific radiation protection program required by 10CFR72.212(b)(6).

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(continued)



~~BASES (continued)~~

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~~ACTIONS~~ ————— A note has been added to the ~~ACTIONS~~ which states that, for this LCO, separate Condition entry is allowed for each ~~TRANSFER CASK~~. This is acceptable since the ~~Required Actions~~ for each ~~Condition~~ provide appropriate compensatory measures for each ~~TRANSFER CASK~~ not meeting the LCO. Subsequent ~~TRANSFER CASKs~~ that do not meet the LCO are governed by subsequent ~~Condition~~ entry and application of associated ~~Required Actions~~.

A.1

If the ~~TRANSFER CASK~~ average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the MPC that did not meet the ~~Functional and Operating Limits~~ in ~~Section 2.0~~. ~~Administrative verification~~ of the MPC fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a mis-loaded fuel assembly is the cause of the out of limit condition. The ~~Completion Time~~ is based on the time required to perform such a verification.

A.2

If the ~~TRANSFER CASK~~ average surface dose rates are not within limits, and it is determined that the MPC was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the ~~OVERPACK~~, once located at the ~~ISFSI~~, would result in the ~~ISFSI~~ offsite or occupational doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72. If it is determined that the out of limit average surface dose rates do not result in the regulatory limits being exceeded, ~~TRANSPORT OPERATIONS~~ may proceed.

B.1

If it is verified that unauthorized fuel was loaded or that the ~~ISFSI~~ offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the transfer cask average surface dose rates above the LCO limit, the fuel

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(continued)

~~BASES~~

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~~ACTIONS~~ ————— ~~B.1 (continued)~~

~~assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid, perform fuel cooldown operations, re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.~~

---

~~SURVEILLANCE~~ ————— ~~SR 3.2.1.1~~  
~~REQUIREMENTS~~

~~This SR ensures that the TRANSFER CASK average surface dose rates are within the LCO limits prior to TRANSPORT OPERATIONS. The surface dose rates are measured on the sides and the top of the TRANSFER CASK at locations described in the SR following standard industry practices for determining average dose rates for large containers. The SR requires specific locations for taking dose rate measurements to ensure the dose rates measured are indicative of the average value around the cask.~~

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~~REFERENCES~~ ————— ~~1. 10 CFR Parts 20 and 72.~~  
~~2. FSAR Sections 5.1 and 8.1.6.~~

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## B 3.2 SFSC Radiation Protection

## B 3.2.2 TRANSFER CASK Surface Contamination

**BASES**

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**BACKGROUND** A TRANSFER CASK is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the TRANSFER CASK may become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the TRANSFER CASK to the ISFSI, or prior to transferring the MPC into the OVERPACK, whichever occurs first, in order to minimize the radioactive contamination to personnel or the environment. This allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.

---

**APPLICABLE SAFETY ANALYSIS** The radiation protection measures implemented during MPC transfer and transportation using the TRANSFER CASK are based on the assumption that the exterior surfaces of the TRANSFER CASKs have been decontaminated. Failure to decontaminate the surfaces of the TRANSFER CASKs could lead to higher-than-projected occupational doses.

---

**LCO** Removable surface contamination on the TRANSFER CASK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm<sup>2</sup> from beta and gamma sources and 20 dpm/100 cm<sup>2</sup> from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the TRANSFER CASK loading process.

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(continued)

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**BASES**

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LCO  
(continued)

Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose. LCO 3.2.2 requires removable contamination to be within the specified limits for the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC. The location and number of surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the MPC means the upper portion of the MPC external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed. The user shall determine a reasonable number and location of swipes for the accessible portion of the MPC. The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC, while implementing sound ALARA practices.

---

APPLICABILITY

The applicability is modified by a note that states that the LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING. This is consistent with the intent of this LCO, which is to ensure loose contamination on the loaded TRANSFER CASK and MPC outside the FUEL BUILDING is within limits. If the MPC transfer is performed inside the FUEL BUILDING the empty TRANSFER CASK remains behind and is treated like any other contaminated hardware under the user's Part 50 contamination control program.

Verification that the surface contamination is less than the LCO limit is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS, when the LCO is applicable. Measurement of surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.

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(continued)

BASES (continued)

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**ACTIONS** A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. Subsequent TRANSFER CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the removable surface contamination of a TRANSFER CASK or MPC, as applicable, that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the TRANSFER CASK or MPC and bring the removable surface contamination within limits. The Completion Time of 7 days is appropriate given that sufficient time is needed to prepare for, and complete the decontamination once the LCO is determined not to be met.

---

**SURVEILLANCE REQUIREMENTS** SR 3.2.2.1

This SR verifies that the removable surface contamination on the TRANSFER CASK and/or accessible portions of the MPC is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification during LOADING OPERATIONS in order to confirm that the TRANSFER CASK or OVERPACK can be moved to the ISFSI without spreading loose contamination.

---

**REFERENCES**

1. FSAR Sections 8.1.5 and 8.1.6.
2. NRC IE Circular 81-07.

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B 3.2 SFSC Radiation Protection Deleted

B 3.2.3 OVERPACK Average Surface Dose Rates Deleted

**BASES**

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**BACKGROUND** — The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions.

---

**APPLICABLE SAFETY ANALYSIS** — The OVERPACK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.

---

**LCO** — The limits on OVERPACK average surface dose rates are based on the shielding analysis of the HI-STORM 100 System (Ref. 2). The limits were selected to minimize radiation exposure to the general public and maintain occupational dose ALARA to personnel working in the vicinity of the SFSCs.

---

**APPLICABILITY** — The average OVERPACK surface dose rates apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. These limits ensure that the OVERPACK average surface dose rates are within the estimates contained in the HI-STORM 100 Topical Safety Analysis Report. Radiation doses during STORAGE OPERATIONS are monitored for the OVERPACK by the SFSC user in accordance with the plant specific radiation protection program required by 10CFR72.212(b)(6).

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(continued)

BASES (continued)

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ACTIONS ————— A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the OVERPACK average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the MPC that did not meet the Functional and Operating Limits in Section 2.0. Administrative verification of the MPC fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a mis-loaded fuel assembly is the cause of the out of limit condition. The Completion Time is based on the time required to perform such a verification.

A.2

If the OVERPACK average surface dose rates are not within limits, and it is determined that the MPC was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the OVERPACK, once located at the ISFSI, would result in the ISFSI offsite or occupational doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72. If it is determined that the out of limit average surface dose rates do not result in the regulatory limits being exceeded, STORAGE OPERATIONS may proceed.

B.1

If it is verified that the correct fuel was not loaded or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the OVERPACK average surface dose rates above the LCO limit, the fuel

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(continued)

BASES

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~~ACTIONS~~ assemblies must be placed in a safe condition in the spent fuel  
~~—(continued)~~ pool. The Completion Time is reasonable based on the time required  
to transfer the MPC back into the TRANSFER CASK, replace the  
transfer lid with the pool lid, perform fuel cooldown operations, re-flood  
the MPC, cut the MPC lid welds, move the SFSC into the spent fuel  
pool, remove the MPC lid, and remove the spent fuel assemblies in an  
orderly manner and without challenging personnel.

---

~~SURVEILLANCE~~ SR 3.2.3.1  
~~REQUIREMENTS~~

This SR ensures that the OVERPACK average surface dose rates are  
within the LCO limits within 24 hours of placing the OVERPACK in its  
designated storage location on the ISFSI. Surface dose rates are  
measured at the locations described in the SR following standard  
industry practices for determining average dose rates for large  
containers.

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- ~~REFERENCES~~
- ~~1.~~ 10 CFR Parts 20 and 72.
  - ~~2.~~ FSAR Sections 5.1 and 8.1.6.
-



### B 3.3 SFSC Criticality Control

#### B 3.3.1 Boron Concentration

##### BASES

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**BACKGROUND** A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where ~~dose rates are measured~~ and the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and ~~vacuum~~ drying is performed. The MPC cavity is backfilled with helium. ~~Then, Additional dose rates are measured~~ and the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. The TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.

For those MPCs containing PWR fuel assemblies of relatively high initial enrichment, credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.

---

**APPLICABLE SAFETY ANALYSIS** The spent nuclear fuel stored in the SFSC is required to remain subcritical ( $k_{\text{eff}} < 0.95$ ) under all conditions of storage. The HI-STORM 100 SFSC is analyzed to stored a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-32 *and* MPC-32F, and for relatively high enrichment PWR fuel loaded in the MPC-24, -24E, and -24EF, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

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(continued)

BASES (continued)

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LCO

Compliance with this LCO ensures that the stored fuel will remain subcritical with a  $k_{\text{eff}} \leq 0.95$  while water is in the MPC. LCOs 3.3.1.a and 3.3.1.b provide the minimum concentration of soluble boron required in the MPC water for the MPC-24, and MPC-24E/24EF, respectively, *for MPCs containing all INTACT FUEL ASSEMBLIES*. The limits are applicable to the respective MPCs if one or more fuel assemblies to be loaded in the MPC had an initial enrichment of U-235 greater than the value in Table 2.1-2 of Appendix B to the CoC for loading with no soluble boron credit.

*LCO 3.3.1.e provides the minimum concentration of soluble boron required in the MPC water for the MPC-24E and MPC-24EF containing at least one DAMAGED FUEL ASSEMBLY or one fuel assembly classified as FUEL DEBRIS.*

~~LCO 3.3.1.ef provides the minimum boron concentration of soluble boron required in the MPC water for the MPC-32 and MPC-32F based on the fuel assembly array/class and the classification of the fuel as a DAMAGED FUEL ASSEMBLY or FUEL DEBRIS. if one or more to fuel assemblies to be loaded had an initial enrichment less than or equal to 4.1 wt.% U-235. LCO 3.3.1.d provides the minimum boron concentration required in the MPC water for the MPC 32 if one or more to fuel assemblies to be loaded had an initial enrichment greater than 4.1 wt.% U-235.~~

All fuel assemblies loaded into the MPC-24, MPC-24E, MPC-24EF, and MPC-32, and MPC-32F are limited by analysis to maximum enrichments of 5.0 wt.% U-235.

*The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes and fuel classifications (intact vs. damaged) are mixed in the same MPC. The ensures the assumptions pertaining to soluble born used in the criticality analyses are preserved.*

(continued)

## BASES

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**APPLICABILITY** The boron concentration LCO is applicable whenever an MPC-24, -24E, -24EF, ~~or -32~~, or -32F has at least one PWR fuel assembly in a storage location and water in the MPC. For the MPC-24 and MPC-24E/24EF, when all fuel assemblies to be loaded have initial enrichments less than the limit for no soluble boron credit as provided in CoC Appendix B, Table 2.1-2, the boron concentration requirement is implicitly understood to be zero.

During **LOADING OPERATIONS**, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water

During **UNLOADING OPERATIONS**, the LCO is applicable when the MPC is re-flooded with water after helium cooldown operations. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is upon entering the Applicability.

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**ACTIONS** A note has been added to the **ACTIONS** which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

### A.1 and A.2

Continuation of **LOADING OPERATIONS**, **UNLOADING OPERATIONS** or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. If the boron concentration of water in the MPC is less than its limit, all activities **LOADING OPERATIONS**, **UNLOADING OPERATIONS** or positive reactivity additions must be suspended immediately.

(continued)

BASES

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ACTIONS  
(continued)

A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately.

One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration be initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours ~~of~~ *prior to* entering the Applicability of the LCO. For LOADING OPERATIONS, this means within four hours of loading the first fuel assembly into the cask.

For UNLOADING OPERATIONS, this means verifying the source of borated water to be used to re-flood the MPC within four hours of commencing re-flooding operations. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

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## BASES

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### SURVEILLANCE REQUIREMENTS

(continued)

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premise of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO

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### REFERENCES

1. FSAR Chapter 6.
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**HI-STORM 100 SYSTEM FSAR**  
**APPENDIX 12.B**  
**COMMENT RESOLUTION LETTERS**  
**(XX Pages Including this Page)**

## CHAPTER 13<sup>†</sup>: QUALITY ASSURANCE

### 13.0 INTRODUCTION QUALITY ASSURANCE PROGRAM

#### 13.0.1 Overview

This chapter provides a summary of the quality assurance program implemented for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the HI-STORM 100 System and HI-TRAC transfer cask designated as important to safety.

~~Table 2.2.6 identifies the structures, systems and components (SSCs) of the HI-STORM 100 System and HI-TRAC transfer cask that are considered important to safety. Table 8.1.6 identifies the ancillary equipment needed for handling and loading operations that has been designated as important to safety.~~

*Important-to-safety activities related to construction and deployment of the HI-STORM 100 System are controlled under the NRC-approved Holtec Quality Assurance Program. Revision 13 of the Holtec QA program manual (Reference [13.0.2]) was submitted to the NRC for review and approval on August 17, 2001 and approved by the NRC on September 25, 2001 (Reference [13.0.4]) under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this previously-approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.*

*The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.*

1. *Holtec Quality Assurance Program Manual*
2. *Holtec Quality Assurance Procedures*
3.
  - a. *Holtec Standard Procedures*
  - b. *Holtec Project Procedures*

*Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type*

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<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the ~~requirements~~-intent 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).

*and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.*

### 13.0.2 Graded Approach to Quality Assurance

*For the HI-STORM 100 System, a graded approach to quality assurance is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Section 13.0.1.*

*NUREG/CR-6407 [13.0.1] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STORM 100 System and HI-TRAC transfer cask. The categories assigned to the cask components are identified in Table 2.2.6. Quality categories for ancillary equipment are provided in Table 8.1.6 on a generic basis. Quality categories for other equipment needed to deploy the HI-STORM 100 System at a licensee's ISFSI are defined on a case-specific basis considering the component's design function.*

*Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 System at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM 100 structures, systems, and components that are important to safety.*

*The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [13.0.3]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536 and has accordingly been added to the list of deviations in Table 1.0.3. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities.*



## 13.1 GRADED APPROACH TO QUALITY ASSURANCE

*This section intentionally deleted.*

For the HI STORM 100 System and HI TRAC transfer cask, a graded approach to quality is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents.

NUREG/CR 6407 [13.1.1] provides descriptions of quality categories A, B and C. These descriptions are provided below:

Category A: ~~Category A items include structures, systems, and components whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.~~

Category B: ~~Category B items include structures, systems, and components whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.~~

Category C: ~~Category C items include structures, systems, and components whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.~~

Using these descriptions along with the quality category assignments from NUREG/CR 6407 [13.1.1], Holtec International has assigned a quality category to each individual component of the HI STORM 100 System and HI TRAC transfer cask. The categories are identified in Table 2.2.6.

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI STORM 100 System on a site specific independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI STORM 100 structures, systems, and components which are important to safety. Regardless of the provisions of the procurement contract, the quality requirements set forth in this document constitute the minimum set of acceptable bases. Activities performed in the course of the previous and ongoing work effort on HI STORM 100 comply with Holtec International's quality assurance program. Holtec International's QA program was developed to meet Nuclear Regulatory Commission (NRC) requirements delineated in 10CFR50, Appendix B, and has been expanded to include provisions of 10CFR71, Subpart H and 10CFR72, Subpart G, for structures, systems, and components designated as

important to safety. A topical report [13.1.2] on the Holtec International QA program has been previously submitted to the NRC. Quality Assurance Program Approval for Radioactive Packages No. 0784 was issued by the NRC. This quality assurance program also applies to the design, material procurement, fabrication, inspection, testing, handling, and repair of the HI-STORM 100 System, including the HI-TRAC transfer cask.

The quality assurance program described in this chapter fully complies with the requirements of 10CFR72 Subpart G, and NUREG-1536 [13.1.3].

## 13.2 PROJECT ORGANIZATION

*This section is intentionally deleted.*

The HI STORM 100 System project has been established under Holtec International's project identification number 5014. This project has been designated as important to safety (ITS), which automatically mandates a rigorously formulated and carefully articulated project management system in accordance with the Holtec Quality Assurance Manual (HQAM). The first requirement of the HQAM is to identify a project team, and to prepare and approve a Project Plan. The HQAM mandates that all activities of an important to safety project be carried out in accordance with the Project Plan. Section 13.3 herein presents the essential elements of the HI STORM 100 project programmatic quality requirements.

The HI STORM 100 project team consists of a project manager, the licensing manager, the QA manager, and a team of technical specialists. A description of Holtec's organizational structure, functions, lines of responsibility, and levels of authority can be found in Holtec Quality Assurance documents.

### 13.3 QUALITY ASSURANCE PROGRAM

*This section intentionally deleted.*

#### 13.3.1 Overview

~~Important to safety (ITS) work on the HI STORM 100 project is performed by Holtec International in accordance with Holtec International's quality assurance program which is designed to satisfy the requirements imposed within 10CFR50 Appendix B, 10CFR71, Subpart H, and 10CFR72, Subpart G. The following provides a summary of Holtec International's quality assurance program implementation to comply with the applicable regulatory requirements.~~

#### 13.3.2 Quality Assurance Program Documents

~~Holtec International's quality assurance program has three levels of controlling documents. The highest level, and overall controlling document, is the Holtec International Quality Assurance Manual (HQAM) which provides the requirements and commitments that Holtec International must follow during the course of any nuclear safety related or important to safety project. The manual is organized into 18 sections that correspond to the eighteen QA program criteria cited in the above referenced regulations.~~

~~The second level of quality assurance program controlling documents is the Holtec International Quality Procedures (HQPs). These procedures provide specific details on how Holtec International implements the requirements and commitments in the quality assurance manual.~~

~~Standard and project specific procedures comprise the third level of quality assurance program controlling documents. These procedures are used to control specific project activities and requirements which are not addressed within the Holtec International quality procedures. Examples of this would be a visual weld examination procedure, liquid penetrant examination procedure, or an in process inspection procedure. These procedures are considered quality assurance records and are controlled in accordance with Holtec International's quality assurance program.~~

#### 13.3.3 Quality Assurance Program Content

~~The requirements and commitments of Holtec International's quality assurance program as specified in the Holtec International quality assurance manual and corresponding quality procedures and project specific procedures (hereafter called quality assurance program documents) are summarized below. Each criterion is summarized separately.~~

## 1. Organization

Holtec International's quality assurance program documents define the quality assurance program related responsibilities of Holtec International personnel, as well as the breakdown of the organizational responsibilities within Holtec International. The Holtec International organization is detailed in the HQAM and HQP 1.0.

Holtec International's quality assurance program requires that the President of Holtec International review the status of the quality program on an annual basis. Furthermore, as part of Holtec International upper management's commitment to Holtec International's quality assurance program, a statement of policy authored by the President of Holtec International is contained in the quality assurance manual. This policy defines Holtec International's commitment to meeting the requirements of 10CFR50, Appendix B; 10CFR71, Subpart H; and 10CFR72, Subpart G, as applicable, on safety related and important to safety projects and also delegates overall responsibility of quality program maintenance to the Quality Assurance Manager. The listing of Structures, Systems, and Components (SSC), defined as important to safety for the HI-STORM 100 System, is provided in Table 2.2.6 of this FSAR.

The Quality Assurance Manager is the person responsible for establishing and maintaining the QA Program. He reports to the Executive Vice President of Holtec International on all quality matters and has the authority and organizational freedom to enforce QA requirements, identify problem areas, recommend or provide solutions to QA problems, and verify the effectiveness of those solutions. As necessary, the Quality Assurance Manager can communicate directly to the President of Holtec International on quality related issues. The minimum qualification requirements for the position of Quality Assurance Manager are contained in the Holtec QA program procedures. Regardless of the education and experience requirements, the QA manager shall be knowledgeable of the applicable codes and standards.

The Quality Assurance Manager has the following typical responsibilities:

- a. Monitor quality issues and keep Management informed of significant conditions adverse to quality.
- b. Initiate, recommend, or provide solutions and verify implementation of corrective actions to nonconforming conditions.

- ~~c. Control or stop further processing, delivery, or installation of a nonconforming item, deficiency, or unsatisfactory condition until proper dispositioning has occurred.~~
- ~~d. Maintain and control the HQAM, HQPs, and standard and project procedures.~~
- ~~e. Review contractual documents to assure inclusion of applicable quality assurance requirements.~~
- ~~f. Interface with clients and regulators during audits.~~
- ~~g. Schedule, perform, and/or oversee audits/surveillances of suppliers of quality related items and services to verify proper implementation of the quality assurance program.~~
- ~~h. Schedule, perform, and/or oversee audits of internal activities to verify compliance with the HQAM.~~
- ~~i. Approve Quality Procedures and Project Plans.~~
- ~~j. Perform periodic reviews of nonconformance reports to identify adverse quality trends for management review and assessment.~~
- ~~k. Coordinate activities to assess the adequacy and effectiveness of the QA program.~~
- ~~l. Schedule and conduct training and indoctrination of personnel performing activities affecting quality.~~
- ~~m. Maintain current qualifications/certifications for personnel performing quality related activities, as appropriate.~~
- ~~n. Maintain a current Approved Vendors List for vendors approved to provide quality related items/services.~~
- ~~o. Maintain a current list of approved computer programs.~~

~~Some of the above listed activities may be performed by personnel designated by the Quality Assurance Manager, although the Quality Assurance Manager retains overall responsibility for assuring proper implementation of the Quality Assurance Program.~~

~~Holtec International may contract with another organization to perform work on important to safety activities. The other organization could be a design agent, manufacturer, supplier, or subcontractor. Any organization performing functions affecting quality of important to safety work must have a QA position with the required authority and organizational freedom, as well as, direct access to upper levels of management. Holtec International shall retain overall responsibility for the QA Program.~~

## ~~2. Quality Assurance Program~~

~~The Holtec International quality assurance program requires that activities important to safety involving design, procurement, fabrication, inspection and testing are performed in accordance with written procedures. Additional project specific procedures are written as needed when specific project requirements are not covered by quality procedures. These additional project specific quality procedures are considered quality assurance records which are controlled in accordance with Holtec International's quality assurance program. QA manuals and procedures, as well as project specific procedures, are controlled and distributed in accordance with the quality assurance program.~~

~~Holtec International personnel performing important to safety activities must be indoctrinated in the Holtec International quality assurance program prior to performing important to safety work in order assure requirements of the QA program are understood. Additionally, a training session is held each year for Holtec International personnel in order to review specific quality assurance requirements. The effectiveness of the quality program is assessed by upper management through annual audits, in-process assessments, and other means.~~

~~Holtec International personnel performing inspection, testing or auditing activities are qualified in accordance with written procedures using guidelines established by the American Society for Nondestructive Testing, American Society of Mechanical Engineers, American National Standards Institute, or other recognized authority, as applicable. These procedures define education, training, experience, and examination requirements for qualifying personnel to perform inspection, testing or auditing. Qualification records are maintained by the quality assurance manager, or designee, and include certification records, bases for qualification, qualification time period, experience and training records, and examination scores, as applicable. Proficiency of qualified personnel shall be maintained as required through retraining, re-examination, and/or re-certification.~~

~~Contractors used by Holtec International to perform important to safety work may have their own quality assurance program which meets or exceeds Holtec International's, or shall perform the work under Holtec International's quality assurance program.~~

~~QA programs of contractors performing important to safety work are reviewed by Holtec's quality assurance organization through audits, assessments, and surveillances to assure applicable QA criteria will be met.~~

~~A project plan is generated for each important to safety project. The project plan contains the necessary information to enable the project team to execute the project in a well-coordinated manner.~~

~~Disputes involving quality which arise from the difference of opinion between personnel from other departments shall be resolved by the QA Manager.~~

### ~~3. — Design Control~~

~~Holtec International's quality assurance program documents establish measures necessary to assure the control of the design process, from input through verification. A design basis is defined in a design specification so that appropriate codes, standards and other relevant documents are used during the course of the design process. Design parameters, as well as miscellaneous design requirements, such as maintenance, repair and storage, are also defined within the Holtec design specification.~~

~~Drawings, procedures and design reports are the three main documents produced by Holtec International through its design process. Holtec International quality program requirements for procedures and drawings are defined in criterion 5 of the HQAM. Measures are established to assure applicable requirements from design bases documents are translated into drawings, procedures, and reports.~~

~~Quality assurance program documents are established to identify and control the authority and responsibilities of all individuals or groups responsible for design reviews and verification activities.~~

~~Holtec International's quality assurance program documents require that all design reports include, as applicable, a defined purpose, assumptions, references, inputs, outputs and results. Design reports are signed by the author and are reviewed by the Project Manager. Additionally, the design report is verified by an individual or group of individuals other than the author of the report. Verification may be made either by qualification testing, design review or alternate calculations. A design verification checklist is used as part of the review process. When qualification testing is used, the prototype shall be subjected to the most adverse design conditions. Surveillances are performed by members of Holtec's Quality Assurance Department to verify that design reports comply with the requirements of Holtec's QA program.~~

~~Measures are established to assure that design verification shall be performed by qualified personnel who did not perform the design analysis. The verifier shall not have influenced inputs or approaches utilized in the analysis. The analyst's supervisor may perform the verification pursuant to the requirements of NQA-1 [13.3.1].~~

~~Holtec International quality assurance program documents require that design verification, if other than by prototype or lead production quality testing, must be satisfactorily completed prior to release for fabrication unless the timing cannot be met. In this case, written justification must be provided to the Quality Assurance Manager or designee and unverified portions of the design must be identified and controlled.~~



~~Changes to a Holtec International design report and specification are subject to the same design controls and must be reviewed and approved in a similar manner to the original.~~

~~Errors in design shall be addressed in accordance with Criteria 15 and 16.~~

~~When applicable, use of commercial items in an important to safety system, structure, or component shall be reviewed for suitability to their intended function.~~

~~Measures are established for the review and disposition of vendor documents including procedures and drawings.~~

~~Measures are established in the QA program to assure valid industry standards and specifications are used in the selection of design inputs (including suitable materials and processes).~~

#### ~~4. Procurement Document Control~~

~~Holtec International's quality assurance program establishes measures to control the preparation, review, approval and issuance of all important to safety purchase orders. Only suppliers approved in accordance with Criterion 7 shall be qualified to supply important to safety items.~~

~~Measures are established within Holtec International's quality assurance program to ensure that purchase orders contain the following information, codes, standards, and specifications, as applicable:~~

- ~~a. a statement of the scope of work to be performed by the vendor;~~
- ~~b. the design basis technical requirements including codes, standards, specifications, etc., to which the item must be designed or manufactured;~~
- ~~c. quality assurance requirements including as applicable, but not limited to, compliance by the vendor with the requirements of 10CFR21 [13.3.2], 10CFR50, Appendix B, 10CFR71, Subpart H, or 10CFR72, Subpart G; and direct reference to the vendor's quality assurance program.~~
- ~~d. permission to gain access to the supplier's or sub-tier supplier's plant facilities and records;~~
- ~~e. identification of documentation required to be supplied by the vendor for approval by Holtec;~~
- ~~f. requirements for reporting and approving disposition of nonconformances;~~
- ~~g. required procedures, tests, and inspections; and~~
- ~~h. record retainage and control requirements.~~

~~All safety significant purchase orders shall be subject to at least one independent review and concurrence. The QA Department shall conduct required surveillances to ensure that safety significant purchase orders are being issued in accordance with the QA program.~~

~~Changes and revisions to purchase orders shall be subjected to the same or equivalent review and approval requirements as the original document.~~

#### ~~5. Instructions, Procedures and Drawings~~

~~Holtec International quality assurance program documents require that activities that are important to safety must be prescribed and accomplished in accordance with written instructions, procedures or drawings. Methods for complying with the 18 criteria set forth within 10CFR50 Appendix B, 10CFR71, Subpart H, and 10CFR72, Subpart G, are also required to be described within defined procedures.~~

~~Instructions, procedures and drawings are required by the Holtec International quality assurance program to include qualitative and quantitative acceptance criteria in order to verify that activities important to safety have been satisfactorily accomplished.~~

~~Measures are established through the Holtec International quality assurance program to prepare, review, approve, and control these instructions, procedures and drawings. The review of these documents is required to be performed by a cognizant verifier other than the author. Revisions to instructions, procedures and drawings are required to be reviewed and approved in a similar manner to the original revision.~~

#### ~~6. Document Control~~

~~Holtec International's quality assurance program documents establish methods to control the review, approval, and issuance of documents and changes thereto, before release, to ensure that the documents are adequate and applicable quality requirements have been incorporated. Documents that must be controlled shall include, but not be limited to: design specifications; design reports; design and fabrication drawings; procurement documents; QA manuals; design criteria documents; and procedures and instructions (i.e., fabrication, inspection, and testing).~~

~~Measures are established in quality assurance program documents to define individuals or organizations responsible for the review, approval, and control of the documents identified above. Document revisions are required to be reviewed, approved, and controlled in a similar manner to the original document. Review of documents is required to be performed by qualified personnel.~~

~~Quality assurance program documents require that documents required to perform a specific activity shall~~

be available at the location where the activity is being performed. Quality assurance program documents also require that obsolete or superseded documents are controlled in order to prevent their inadvertent use.

An index of project documents is maintained in order to allow identification of the latest revision of applicable documents. This list includes, but is not limited to, design reports, specifications, procedures, and drawings.

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7. Control of Purchased Material, Equipment and Services

Holtec International quality assurance program documents define measures to ensure that important to safety materials, equipment and services conform to procurement documents. Procedures are established to define requirements for procurement document control, supplier evaluation and selection, vendor surveillance, and receipt inspection in order to assure purchased items are properly controlled from the procurement phase through item receipt.

Holtec International quality assurance program documents require that Holtec International qualified personnel evaluate Holtec International subcontractors supplying important to safety items and services prior to contract award. A vendor shall be evaluated to determine its technical capability as well as its production capability. Those vendors found to have satisfactory technical and production capabilities are submitted to the quality assurance department for a quality assurance evaluation. The quality assurance evaluation, which shall be documented, shall assess past performance and also determine the capabilities of the vendor to comply with required codes and QA criteria through audit, surveillance, or other source evaluation, as applicable. Unacceptable conditions discovered by Holtec International quality assurance are addressed through nonconformances and audit findings, as applicable. Holtec International shall impose its own quality assurance program on vendors which are determined not to have an adequate quality assurance program; or shall require changes in the supplier's quality assurance program to make it acceptable to Holtec International; or shall perform dedication of the items through surveillance, inspections, and tests in accordance with Holtec International's QA program, as applicable. Qualified suppliers of important to safety items, equipment, and services must be placed on Holtec International's Approved Vendors List. Specific requirements for placing vendors on the Approved Vendor List are defined within Holtec International quality assurance program documents. As applicable, this includes an audit, surveillance, or other source evaluation of the vendor to verify QA program conformance to applicable codes and implementation of the QA program. Measures for performing audits, surveillances, and other source evaluations are defined in quality assurance program documents. As applicable, the QA program requires triennial audits, surveillances, or other source evaluations in order to verify continued implementation of their QA program and maintenance on the Approved Vendors List.

Measures for performing supplier surveillances are defined within Holtec International quality assurance program documents. Source surveillance is used to determine that in process work is being

~~performed by the supplier in accordance with purchase order requirements. The Project Manager, in conjunction with the Quality Assurance Manager, must determine the extent of source surveillance required for a particular job or supplier based on the important to safety classification, complexity of the item, and quantity. Holtec International quality assurance program documents define types of surveillance activities that may be performed including hold point verification. Project specific procedures and procurement documents define, when applicable, necessary inspection points to be performed by Holtec, and inspection and test acceptance criteria.~~

~~Measures for performing receipt inspection activities are defined within Holtec International quality assurance program documents. Receipt inspection is performed in order to verify received items meet the requirements of the purchase order. The extent of receipt inspection to be performed on vendor furnished items in order to assure items are properly identified and conform to purchase order requirements is established through Holtec International quality and project procedures, or vendor procedures approved by Holtec. Inspection records, material test reports, and/or certificates of conformance attesting to the acceptance of the item are reviewed, as applicable, for acceptability as part of the receipt inspection process. When item acceptance is contingent on post-installation testing or inspection, the acceptance criteria shall be defined with vendors through procurement documents prior to item use. Items and materials that have completed receipt inspection and are released for fabrication or further use are controlled in accordance with quality assurance program documents.~~

~~Measures have been established through Holtec International quality assurance program documents to control items discovered during receipt inspection to have a nonconforming condition. These measures include segregation and identification of items, evaluation of the nonconforming items, and disposition with justification, as required.~~

~~Holtec International quality assurance program documents establish measures to assure that a supplier provides the documentation for a received part as required by the purchase order. These documents include, but are not limited to, material test reports, inspection and test reports, certificates of conformance and nonconformance reports, as applicable. Review of these documents for conformance to procurement documents is required.~~

#### ~~8. Identification and Control of Materials, Parts and Components~~

~~Holtec International quality assurance program documents establish measures to ensure that materials, parts and components, including partially fabricated assemblies, are adequately identified and controlled in order to preclude the use of incorrect or nonconforming items. Measures are established by Holtec International through its quality documents to ensure that limited life items are controlled in order to preclude their use once the shelf life of these items has expired.~~

~~Measures are established by Holtec International through quality assurance program documents in order to provide the means for material, part or component identification so that items maintain traceability to~~

appropriate documentation such as drawings and test reports throughout fabrication, installation and use, and to preclude use of incorrect or defective items. Markings are required to be made such that they are not detrimental to the item. Any specific identification or marking requirements are identified through drawings, procedures, or specifications.

#### 9. Control of Special Processes

Holtec International quality assurance program documents establish measures to ensure that special processes such as welding, lead pouring, neutron shield material installation, and NDE examinations are controlled. Specific special processes are typically identified in fabrication specifications. Procedures, equipment, and personnel used to perform special processes are required to be qualified in accordance with applicable codes, standards and specifications. Special process operations shall be performed by appropriately qualified personnel using written and approved procedures, as applicable. Special process operations are required to be documented and verified. Special process records including procedure, equipment and personnel qualifications, as well as special process operation results are required to be maintained as quality records.

#### 10. Licensee Inspection

Inspections are required to be performed in accordance with written procedures in order to verify conformance of quality affecting activities. Drawings and specifications are used in conjunction with the procedures to define specific acceptance criteria. Inspection procedures include, as applicable, identification of characteristics and activities to be inspected, acceptance and/or rejection criteria, methods of inspection, identification of the individuals or groups responsible for performing the inspection operation, recording of inspection results, identification of hold and witness points, approval requirements for inspection data and inspection prerequisites such as personnel qualifications. Inspection results are documented and signed by the applicable inspector. Inspections through sampling shall use known standards as applicable for the basis of acceptance.

Measures are established within Holtec International quality assurance program documents to ensure that structures, systems, and components important to safety are, upon receipt, inspected to verify that the item meets purchase order requirements. Control of materials, both before and after receipt inspection, are defined for both accepted and nonconforming material within Holtec International quality assurance program documents.

Measures for in-process control are established through project-specific procedures for situations when direct inspection would be impractical. In-process controls when required, may include, but are not limited to, monitoring of processing methods, equipment and personnel, as well as review of in-process documentation.

Measures are established within the quality assurance program documents to assure that reworked or repaired items are inspected to the original requirements, or approved deviation and new

requirements.

~~Holtec International quality assurance program documents establish measures to ensure that nonconformances identified during the course of fabrication are resolved prior to, or during final inspection; that items which are inspected must be identifiable and traceable to specific records; and that inspection records must be reviewed by the Holtec International QA Manager, or designee, to verify the inspection requirements have been satisfied.~~

~~Holtec International quality assurance program documents require that inspectors shall be qualified in accordance with applicable codes and standards and shall be properly trained. Inspector qualification records are maintained within the quality assurance files and are required to be kept current. Measures are defined within Holtec International quality assurance program documents to ensure that inspection personnel are independent from personnel performing the activity being inspected.~~

#### 11. Test Control

~~Holtec International quality assurance program documents establish measures to ensure that applicable test programs (i.e., load tests, leak tests, hydrostatic tests, production tests, etc.) are performed in accordance with written procedures, as applicable. Test procedures include, as applicable: test equipment and calibration requirements; material requirements; personnel qualifications; prerequisites (including environmental conditions); detailed performance instructions; hold points; acceptance and rejection criteria; instructions for documenting and evaluating results; and documentation approval requirements.~~

~~The acceptance test program is defined in Chapter 9 of the FSAR for the HI-STORM 100 System and shall be implemented for each system to verify that SSCs conform to the specified requirements and will perform satisfactorily in service.~~

~~Only qualified personnel shall evaluate test results for acceptability.~~

#### 12. Control of Measuring and Test Equipment

~~Holtec International quality assurance program documents establish measures to ensure that measurement and test equipment shall be calibrated, adjusted and maintained at prescribed intervals or prior to use. Calibrations are required to be performed in accordance with written procedures or standards. Measuring and test equipment is required to be controlled such that the next calibration date and traceability back to calibration records is maintained.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that calibrations of measuring and test equipment are performed using calibration standards that are both traceable and have known valid relationships to nationally recognized standards. When no known~~

~~recognized standard exists, the basis for the calibration is required to be defined and documented.~~

~~Measures are established within Holtec International quality assurance program documents to control measuring and test equipment which is found to be out of calibration. These controls include validation of all previous inspection and test results from the time the item was found to be out of calibration back to the time of the previous acceptable calibration of the same item. Measuring or test equipment found to be out of calibration is required by Holtec International quality assurance program documents to be repaired and recalibrated prior to next use, or replaced.~~

~~A master list of calibrated tools and equipment is required to be kept in order to maintain a complete calibration status of each item.~~

### ~~13. Handling, Storage and Shipping~~

~~Holtec International quality assurance program documents establish measures to ensure that cleaning, handling, storage and shipping of items are accomplished in accordance with design requirements to preclude damage, loss, or deterioration by environmental conditions. These activities are performed in accordance with written instructions or procedures as necessary. Measures for establishing provisions for the use of special handling, lifting or storage equipment in order to adequately identify and preserve items, components or assemblies are provided within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that a review of packaging be performed prior to item shipment in order to assure packaging meets approved drawings, specifications and codes. Additionally, verification of completion of documentation, including procedures, manuals and inspection and test results is required to be performed prior to shipment. Physical identification of the item shall be verified prior to shipment.~~

~~14. Inspection, Test and Operating Status~~

~~Holtec International quality assurance program documents establish measures to ensure the inspection, test and operating status of items is known by organizations responsible for quality activities.~~

~~Measures are established by Holtec International through its quality assurance program documents to control the application and removal of status indicators such as markers and tags. Additionally, Holtec International quality assurance program documents establish measures to ensure that if required operations such as tests or inspections are bypassed, such action is taken through controlled procedures and under cognizance of the quality assurance department.~~

~~Controls on nonconforming items are summarized in Criterion 15.~~

~~15. Nonconforming Materials, Parts or Components~~

~~Holtec International quality assurance program documents establish measures to ensure control of nonconforming important to safety items, services, and activities. This includes provisions for the identification, documentation, tracking, segregation, review, disposition of nonconforming items, and notification of the affected organizations, as appropriate.~~

~~Holtec International quality assurance program documents establish measures to ensure that nonconforming items, services or activities shall be reviewed and dispositioned. Provisions are included to ensure that nonconforming services or activities, including those of suppliers, for which the recommended disposition is "accept as is" or "repair", shall be submitted to the client for approval, if required.~~

~~Measures are established within Holtec International quality assurance program documents to require nonconformances to be identified through deviation reports and corresponding corrective actions (which may include repair, rework, and inspection requirements). Individuals responsible for review and disposition of nonconforming items are identified within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to control further processing, delivering, or installation of nonconforming or defective items pending a decision on its disposition. Measures are established through Holtec International quality assurance program documents to ensure that nonconforming items are segregated and controlled until proper disposition is completed.~~



~~Holtec International quality assurance program documents establish measures to ensure that the acceptability of nonconforming items is verified by inspecting or testing the nonconforming item against original requirements after designated repair or rework. Final disposition of nonconforming items shall be defined and documented.~~

~~Measures are established within Holtec International quality assurance program documents to permit anyone who discovers a nonconformance to report it in accordance with quality assurance program documents. Provisions are established to ensure that nonconformances are evaluated for the purpose of determining if reporting pursuant to 10CFR21 [13.3.2] is required.~~

~~Holtec International quality assurance program documents require that nonconformances be assessed by the Quality Assurance Manager on a defined basis to determine any quality trends. Any trends or significant results shall be evaluated by appropriate management personnel for development of correction actions.~~

~~Nonconformance reports are considered part of the quality records package. As built conditions are required to be documented as applicable.~~

#### ~~16. Corrective Action~~

~~Holtec International quality assurance program documents establish measures to ensure that causes of conditions adverse to quality are promptly identified and reported to upper management through deviation reports and corrective action reports. Measures are also established to ensure that corrective actions are performed on identified nonconforming conditions or items, and that follow-ups are performed and documented as applicable to verify implementation and effectiveness of the corrective action.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that follow up activities are performed to verify that corrective actions have been correctly implemented so as to minimize the possibility of recurrence of the nonconforming condition. Individuals responsible for verifying and documenting corrective action are identified within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to document and evaluate significant conditions adverse to quality through root cause evaluations. These evaluations are performed by qualified individuals and reviewed by cognizant levels of management.~~

## 17. Quality Assurance Records

~~Holtec International quality assurance program documents require that evidence of activities affecting quality shall be documented and shall provide sufficient information to permit identification of the record with the items or activities to which it applies. Quality assurance records include, but are not limited to, design, procurement, manufacturing and installation records; audits (internal and external); nonconformance reports; inspection and test results; drawings (including as-built) and specifications; analysis reports (i.e., failure, seismic, etc.); personnel qualifications and training (including retraining) records; procedures (i.e., inspection, testing, calibration, etc.); calibration records; equipment qualification; corrective action reports; operating logs and completed travelers; material test reports; and design review documents.~~

~~Holtec International quality assurance program documents require that inspection and test records shall, as applicable, contain observations, evidence of inspection or test performance, results of inspections or tests, names of inspectors, date of tests, test personnel and data recorders, equipment identification, and evidence of acceptability. Any nonconforming conditions shall be addressed in accordance with Criterion 15.~~

~~Holtec International quality assurance program documents establish measures to ensure that documents defined as quality assurance records are legible and that they reflect the total of work performed.~~

~~Quality assurance records are defined as either "lifetime" or "nonpermanent", as appropriate. Holtec International quality assurance program documents define which quality assurance records are "lifetime" and which are "nonpermanent". "Lifetime" records are those records that pertain to the design, fabrication and installation of a particular item such that the records can demonstrate the capability of the item and provide evidence of activities supporting the acceptability of the item. These records demonstrate the capability for safe operation; provide evidence of repair, rework, replacement or modification; aid in determining the cause for an accident or malfunction of an item; or provide a baseline for inservice inspection. Examples of "lifetime" records include design reports, drawings, procedures and inspection reports. "Nonpermanent" records are those records that show evidence of an activity being performed but do not meet the criteria for "lifetime" records. Examples of "nonpermanent" records include document transmittal forms and surveillance reports. "Nonpermanent" record retention times are defined within Holtec International quality assurance program documents.~~

~~Holtec International quality assurance program documents establish measures to ensure quality assurance records are properly controlled from receipt through long term storage. Responsibilities for receipt, storage, retrieval and disposal of quality assurance records are provided within Holtec International quality assurance program documents. Records are required to be indexed so that they are readily retrievable.~~

~~Holtec International quality assurance program documents define storage requirements in order to assure quality assurance records are not damaged or destroyed. Quality assurance records are required to be stored in boxes, cabinets or shelves, or on the electronic network, and shall be protected from such conditions as water, fire, etc. Measures are established through Holtec International quality assurance documents to ensure records requiring special storage requirements are stored properly. Quality assurance record storage areas are required by Holtec International quality assurance program documents to have controlled access. In the case where a quality assurance record is damaged or lost, it is required to be replaced immediately in a controlled manner by responsible personnel.~~

#### 18. Audits

~~Holtec International quality assurance program documents define a comprehensive audit program including independence of the auditors from the area being audited, audit schedule requirements, identification of auditors and their required qualifications, access provisions for audit personnel, documentation requirements, methods for reporting audit findings, and methods for corrective actions and follow-ups.~~

~~Holtec International quality assurance program documents require that schedules be defined for internal and external audits. Audit plans are required to be written for each audit and shall define the key activities or areas to be audited.~~

~~Audits are performed in accordance with written procedures and/or checklists. Audits are performed in order to provide a comprehensive independent verification and evaluation of procedures and activities affecting quality, and to verify and evaluate a supplier's QA program, procedures, and activities. As appropriate, audit teams may contain members who are technical experts in the areas being audited. Holtec International internal audits are required to be performed annually and shall review all aspects of Holtec International's quality assurance program in order to determine the effectiveness of the program. External audits are performed per Criterion 7, as necessary, and shall evaluate all applicable and Holtec International relevant portions of the vendor's quality assurance program.~~

~~Holtec International quality assurance program documents establish qualification requirements for auditors including lead auditors. Additionally, responsibilities of audit personnel regarding the performance of the audit as well as the follow up documentation (i.e., audit report, findings etc.) are defined within the same documents.~~

~~The Holtec International quality assurance program documents establish requirements for the performance of pre and post audit conferences. The pre audit conference is used to define the scope of the audit as well as the specific areas to be audited, and define a schedule and agenda for the audit. The post audit conference is used to discuss the results of the audit with the audited party.~~

~~Holtec International quality assurance program documents establish measures for writing of audit reports and provide instructions for the processing of findings and their corresponding corrective actions. Corrective action responses are required to clearly state the corrective action taken to correct the nonconforming condition and date of implementation. Audit reports shall be transmitted to responsible personnel at the audited organization for review and implementation of corrective actions, when required. Reports of internal audits shall be transmitted to the president of Holtec International.~~

~~Holtec International quality assurance program documents require that the audit team verify that corrective action responses are made in a timely manner, that the corrective action responses are adequate, and that corrective actions have been properly implemented.~~

13.4            PROJECT PLAN

Section deleted.

*This section intentionally deleted.*

~~The structure of the Holtec International organization and the assignment of responsibilities for each activity ensures that the designated responsible parties will perform the necessary work to achieve and maintain the quality requirements specified in the HQAM. Conformance to established requirements will be verified by individuals and groups not directly responsible for the performance of the work. The QA Manager, who directly reports to the Executive Vice President of Holtec International, has been designated as the party responsible for verifying quality, and he has the required authority and organizational freedom, including independence from influence of cost and schedule, to effectively complete his responsibilities. The QA Manager can also communicate directly to the President of Holtec International regarding quality assurance activities.~~

~~The Holtec International Quality Assurance Program is documented in the HQAM, HQPs and project specific procedures, and provides adequate control over activities affecting quality, as well as structures, systems, and components that are important to safety, to the extent consistent with their relative importance to safety. The QA program describes a management system and controls, that when properly implemented, will comply with the requirements of Subpart G to 10CFR Part 72 and 10CFR Part 21 [13.3.2].~~

~~Design analyses and engineering documentation for the thermal, structural, confinement, criticality, shielding, and operational capabilities of the HI-STORM 100 System for normal, off-normal and postulated accident conditions are carried out in accordance with the 18 criteria in the HQAM. In addition, those activities and items designated as important to safety and related to the material specification and procurement for the HI-STORM overpack and MPC canister, as well as the HI-STORM 100 lifting equipment, are subject to Holtec QA program procedures. Governing procedures include those for procurement document control, control of purchased items and services, material handling, and instructions and drawings which control material requirements.~~

~~Further, the fabrication, testing and inspection of the HI-STORM 100 System by Holtec International and its subcontractors will be conducted in accordance with all QA program requirements, including those activities and project procedures addressed by the 18 criteria, especially those covering design control, identification, and control of materials, parts and components, test control, inspection procedures, control of special processes, control of measuring and test equipment, and inspection and test status documentation.~~

~~The operation, maintenance, repair and modification of the HI-STORM 100 System will be governed by the licensee's (e.g., utility) QA program with support and record maintenance as required by Holtec's QA program and regulatory requirements. These activities will be verified and audited on a periodic basis with respect to control of nonconforming materials, parts or components, corrective action, quality assurance~~

records, audits, and reviews of ongoing inspections, surveillances, and operating status.

In conclusion, the Holtec International QA Program complies with the applicable NRC regulations and industry standards, and will be implemented for the HI-STORM 100 dry cask storage system.

13.6            REFERENCES

- [13.40.1]        NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.
- [13.40.2]        Holtec International Quality Assurance Program, *Revision 13*. ~~Topical Report for 10CFR71, Subpart H and 10CFR72, Subpart G, Holtec International Report HI 941152, Rev. 2 (8/4/94).~~
- [13.40.3]        NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.
- [13.0.4]        *NRC QA Program Approval for Radioactive Material Packages No. 0784, Revision 3.*
- [13.3.1]        ~~NQA 1, "Quality Assurance Program Requirements for Nuclear Facilities"~~
- [13.3.2]        ~~U.S. Code of Federal Regulations, Title 10, "Energy", Part 21, "Reporting of Defects and Noncompliance."~~



**HI-STORM FSAR**

**APPENDIX 13.A**

**~~DESIGN VERIFICATION CHECKLIST~~**

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**~~APPENDIX 13.A CONTAINS A TOTAL OF 10 PAGES, INCLUDING THIS PAGE~~**

*APPENDIX 13.B*

*INTENTIONALLY DELETED*

**HI-STORM FSAR – APPENDIX 13.B**

**HOLTEC QA PROCEDURES**

<b>PROCEDURE NUMBER</b>	<b>TITLE OF PROCEDURE</b>	<b>10CFR72 SUBPART G QA CRITERIA</b>
1.0	Organization and Responsibilities	1
2.0	Quality Assurance Program	2
2.1	Quality Assurance Manual and Procedure Control	
2.2	Execution of HQAM and Extension to a Fabricator's Facility	
2.3	Quality Forms for Quality Assurance Program Implementation	
2.4	Quality Assurance Requirements for 10CFR71 and 10CFR72	
2.5	Quality Assurance Requirements for Supply of ASME Section III Materials, Components, and Equipment	
2.6	Execution of Quality Requirements and Extension to a Fabricator's Facility for Important to Safety Categories B and C Items.	
3.0	Contract Administration and Design Control	3
3.1	Design Input Requirements	
3.2	Design Analysis	
3.3	Design Verification	
3.4	Design Specifications and Design Criteria Documents	
4.1	Purchase Orders	4
4.2	Material Purchase Specifications	
5.1	Engineering Drawings	5
5.3	Standard and Project Procedures	
5.4	Fast-Fax Analysis	

## HI-STORM FSAR – APPENDIX 13.B

### HOLTEC QA PROCEDURES

PROCEDURE NUMBER	TITLE OF PROCEDURE	10CFR72 SUBPART G QA CRITERIA
6.0	Document Control	6
6.1	Project Document Transmittal and Control	
6.2	Document Classification	
7.0	Receipt Inspection	7
7.1	Supplier Selection	
7.2	Supplier Surveillance	
7.3	Material Dedication – Steel and Weld Wire (Excluding Section III Material)	
7.4	Approved Vendor List	
7.5	Material Dedication Procedure (For Items Not Covered by HQP 7.3)	
7.6	Sampling Plan	
8.0	Material and Item Identification and Control	8
9.0	Qualification of Personnel Performing Holtec Special Processes	9
9.1	Written Practice for Qualification of NDE Personnel	
9.2	Welder Qualification Requirements	
9.3	Inspector Qualification for Non-NDE Activities	
11.0	Computer Programs (Formerly HQP 5.2)	11
12.0	Equipment Calibration and Control of Measuring and Test Equipment	12
14.0	Inspection and Test Status	10,11,14
15.1	Reporting of Defects and Noncompliances per 10CFR21	15
15.2	Nonconformances	

**HI-STORM FSAR – APPENDIX 13.B**

**HOLTEC QA PROCEDURES**

<b>PROCEDURE NUMBER</b>	<b>TITLE OF PROCEDURE</b>	<b>10CFR72 SUBPART G QA CRITERIA</b>
16.0	Corrective Action (formerly Non-Conformance and Corrective Action)	16
16.1	Root Cause Evaluations	
17.0	Quality Assurance Records	17
18.1	Certification of Audit Personnel	18
18.2	Audits	
19.1	Personnel Reliability Program	2
19.2	Field Services	
19.3	Qualification Requirements and Duties of Registered Professional Engineers for Section III, Division 1 Certifying Activities	

Notes: 1. Handling, Storage, and Shipping Requirements are specified in the QA Manual (Section XIII). These activities are performed by Holtec subcontractors in accordance with project specific procedures.