HOLTEC INTERNATIONAL

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

Telephone (856) 797-0900
Fax (856) 797-0909

February 20, 2004

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

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 2
Supplemental Response to Request for Additional Information 2

References:
1. Holtec Project 5014
2. USNRC Letter, Christopher Regan, to Holtec, B. Gutherman, "Request for Additional Information for the Holtec International HI-STORM 100 Amendment 2," dated January 13, 2004.
3. Holtec Letter, Brian Gutherman, to USNRC Document Control Desk, "Response to Request for Additional Information 2," dated February 2, 2004.

Dear Sir:

As committed in the Reference 3 letter, we herewith provide Supplement 2 to License Amendment Request (LAR) 1014-2, Revision 2. This supplement provides the modified proposed changes to the HI-STORM CoC and FSAR required to support our responses to the second round RAI pertaining to this amendment request. In addition, as previously discussed, a small number of editorial corrections not related to the RAI are also provided for clarification. All changes submitted with this supplement are summarized in Attachment 1. The following attachments are provided. Enclosed with this letter are instructions for updating the documents in the LAR notebook.

Attachment 1: Tabular summary of changes provided with this LAR supplement.

Attachment 2: Summary of Proposed Changes: This document has been appropriately revised to reflect the modifications in the submittal made in response to the NRC's second round RAI.

Attachment 3: Replacement mark-up pages for proposed CoC changes.

Attachment 4: Replacement revised pages for proposed CoC changes.

Attachment 5: Replacement pages for proposed FSAR changes. If there was any change to a Proposed Revision 2, 2A, or 2B FSAR Section, either individual pages or the entire section is provided with "Proposed Revision 2C" in the footer. New Appendix 2.C, including Figure 2.C.1 is identified similarly. The current revision level of each

NMSS01
A053

HOLTEC INTERNATIONAL

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

Telephone (856) 797-0900
Fax (856) 797-0909

FSAR page and figure is provided in an 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, except for FSAR Sections 4.4, 4.5, and 7.1. FSAR Sections 4.4, 4.5, and 7.1 show only the revised text in italic font (i.e., no strikeouts are indicated).

We appreciate the SFPO's continued diligent review of this amendment request and look forward to receiving the amended CoC in the near future. Please contact the undersigned if you require additional information.

Sincerely,

Brian Gutherman, P.E.
Manager, Licensing and Technical Services

Approved:

K.P. Singh, Ph.D, P.E.
President and CEO

Document ID: 5014507

Attachments: As Stated

cc: Mr. Christopher Regan, USNRC (w/hard copy attachments and CD of compiled LAR)
HUG Licensing Correspondence Distribution (CD of compiled LAR)
Holtec Groups 1, 2, and 4 (w/o attach.)

CHANGES TO LAR 1014-2 IN REVISION 2, SUPPLEMENT 2

CHANGE	AFFECTED LAR SECTION	DESCRIPTION OF CHANGE
1	CoC, Appendix A, LCO 3.1.4 and associated Bases in FSAR Appendix 12.A.	New LCO to add TS requirements for the Supplemental Cooling System.
2	CoC, Appendix A, Table 3-1.	Modified this table to reflect different drying requirements to coincide with new information from NRC (1/29/04 memo Brown to Hodges).
3	CoC Appendix A, Section 5.7	Added Subsection 5.7.4 to add specific dose rate limits for the overpack and re-numbered subsequent subsections.
4	CoC, Appendix B, Section 2.1.2	Changed "fuel cell location" to "fuel storage location" for consistency.
5	CoC Appendix B, Table 2.1-1 and Section 2.4	Modified language throughout to refer to PWR decay heat limits per fuel storage location rather than fuel assembly. This change is not required for the BWR MPC models because there is no non-fuel hardware associated with BWR fuel (fuel channels are not considered NFH).
6	CoC Appendix B, Subsection 2.4.3.7	Re-numbered Subsection 2.4.3.7 as 2.4.4.
7	CoC Appendix B, Section 3.2.7 and FSAR Section 1.2.1.3.1.2	Increased maximum boron carbide content in METAMIC from 32.5 to 33.0 weight percent.
8	CoC, Appendix B, New Section 3.2.8	New section added to incorporate FSAR Subsection 9.1.5.3 by reference.
9	CoC Appendix B, Section 3.4.10 and Table 3-2a	Deleted this subsection and table in lieu of new LCO 3.1.4.
10	FSAR Table 1.0.2 and Section 2.0.1	Changed revision of ISG-11 from 2 to 3.
11	FSAR Tables 1.0.3, 1.D.1, and 2.0.2, and Section 2.0.2.	Changed basis for increasing overpack concrete temperature limit from NUREG-1536 back to ACI 349, Appendix A with additional justification.
12	FSAR Section 1.1	Editorial change in paragraph 8 to remove "100" from the 2nd sentence. The sentence applies to all overpack designs.
13	FSAR Subsection 1.2.1.3.1.1	Deleted italics from discussion pertaining to Boral and hydrogen. This information was added to the FSAR under 10 CFR 72.48 in response to an NRC violation.

CHANGES TO LAR 1014-2 IN REVISION 2, SUPPLEMENT 2

14	FSAR Subsection 1.2.1.3.1.2	Changed "32.5%" to "33.0" in the 10th paragraph to provide more fabrication flexibility. Deleted "Reference [1.2.14]" after "other's work" in the last paragraph (reference did not exist).
15	FSAR Subsection 1.2.2.2	Added paragraph to address use of the Supplemental Cooling System (SCS) (two places).
16	FSAR Section 1.6	Added NUREG/CR-5661 as new Reference 1.2.14.
17	FSAR Sections 2.0.1 and 2.0.3, and Table 2.0.1	Changed revision of ISG-11 from 2 to 3 and modified text to discuss normal short term operating temperature for low burnup fuel being 1058°F and use of the SCS.
18	FSAR Subsection 2.1.9.1	Modified language throughout to refer to decay heat limits per fuel storage location rather than fuel assembly.
19	FSAR Sections 2.2 and 2.2.1.5, and Table 2.2.3 (second footnote)	Changed the revision level of ISG-11 from 2 to 3 in the definition of Short-Term Operations. Addressed moderate burnup fuel temperature limit during short term normal operations and added reference to NRC's 1/29/04 Brown-to-Hodges memo.
20	FSAR Tables 2.2.13 and 2.2.14	The nomenclature for internal MPC pressure has been modified to recognize the difference between the normal and off-normal pressure limits.
21	FSAR Subsection 2.3.2.2	Deleted extra line break in text.
22	FSAR Section 2.6	Changed the revision level of ISG-11 from 2 to 3 in Reference [2.0.8] and added NRC Brown-to-Hodges memo as Reference [2.0.9].
23	FSAR, New Appendix 2.C	Appendix 2.C has been created to describe the SCS system and define its design criteria.
24	FSAR Sections 4.0, 4.1, 4.3, 4.4, 4.5, 4.6, and 4.7	Chapter 4 was modified throughout to reflect ISG-11, Revision 3 and the January, 2004 PNNL white paper pertaining to low burnup fuel cladding hoop stress.
25	FSAR Appendix 4.B	Appendix 4.B was modified to provide additional detail on thermal margins.
26	FSAR Section 8.1 and 8.3, and Table 8.1.6	Chapter 8 was modified to address the use of the Supplemental Cooling System.
27	FSAR, Subsection 9.1.5.3	Added a note preceding Subsection 9.1.5.3 stating that this section of the FSAR is incorporated by reference into the CoC and cannot be changed without prior NRC approved via CoC amendment. Also changed font of the entire subsection to bold to highlight this restriction.

CHANGES TO LAR 1014-2 IN REVISION 2, SUPPLEMENT 2

28	FSAR Subsection 10.3.2	Subsection 10.3.2 was modified to clarify the dose estimates for surveillance and maintenance.
29	FSAR Section 11.1	New off-normal pertaining to the FHD System and Supplemental Cooling System (SCS) have been added.
30	FSAR Section 11.2	New accident event pertaining to the SCS has been added.
31	FSAR, Tables 12.1.1 and 12.1.2	Added new LCO 3.1.4 to these tables.
32	FSAR, Subsection 12.2.1, item 11	Added supplemental cooling to sixth bullet.
33	FSAR, Subsection 12.2.2	Removed "leakage testing" from item 8 and added a new item 12 to address the supplemental cooling system in the dry run training program.
34	FSAR, Subsection 12.2.10	Replaced "fuel assembly" with "approved cask contents."
35	FSAR, Appendix 12.A	Revised TOC to add Bases 3.1.4, restored the last sentence in the LCO section of Bases 3.1.2, added reference to LCO 3.1.4 in the last sentence of the bases for Required Action B.2.2, and added new Bases 3.1.4 to support new LCO 3.1.4 in the CoC.

INSTRUCTIONS FOR LAR 1014-2, REVISION 2, SUPPLEMENT 2

1. Insert the cover letter and Attachment 1 in the front of the LAR notebook.

2. Replace the Summary of Proposed Changes in its entirety.

3. Remove the following markup CoC pages and replace/insert the enclosed modified markup CoC pages:
   a. Replace Appendix A, Table of Contents (page i)
   b. Insert new LCO 3.1.4 in Appendix A after existing LCO 3.1.3 (page 3.1.4-1/2)
   c. Replace Appendix A, Table 3-1 (page 3.4-1)
   d. Replace Appendix A, pages 5.0-5 through 5.0-7 with new pages 5.0-5 through 5.0-8
   e. Replace Appendix B, pages 2-1/2, 2-9/10, 2-27 through 2-34, 2-39 through 2-46, and 2-63 through 2-66
   f. Replace Appendix B, Section 3.0, "Design Features" in its entirety (21ages)

4. Remove the following revised CoC pages and replace/insert the enclosed modified revised CoC pages:
   a. Replace Appendix A, Table of Contents (page i)
   b. Insert new LCO 3.1.4 in Appendix A after existing LCO 3.1.3 (page 3.1.4-1/2)
   c. Replace Appendix A, Table 3-1 (page 3.4-1)
   d. Replace Appendix A, pages 5.0-5/6 with new pages 5.0-1 through 5.0-7
   e. Replace Appendix B, pages 2-1/2, 2-9/10, 2-27 through 2-34, 2-39 through 2-46, and 2-63 through 2-66
   f. Replace Appendix B, Section 3.0, "Design Features" in its entirety (21 pages)

5. Remove the FSAR Table of Contents and List of Effective Pages, Revision 2B in their entirety and replace with FSAR Table of Contents and List of Effective Pages, Rev. 2C.

6. Remove FSAR page 1.0-13/14, Proposed Rev. 2B and replace with enclosed FSAR page 1.0-13/13, Proposed Rev. 2C.

7. Remove FSAR pages 1.0-25 through 1.0-30, Proposed Rev. 2B and replace with enclosed FSAR page 1.0-25 through 1.0-30, Proposed Rev. 2C.

8. Remove FSAR page 1.1-1/2, Proposed Rev. 2B and replace with enclosed FSAR page 1.1-1/2, Proposed Rev. 2C.

9. Remove FSAR page 1.2-3/4, Proposed Rev. 2B and replace with enclosed FSAR page 1.2-3/4, Proposed Rev. 2C. Do not remove the figures from this section.

10. Remove FSAR pages 1.2-13 through 1.2-18, Proposed Rev. 2B and replace with enclosed FSAR pages 1.2-13 through 1.2-18, Proposed Rev. 2C. Do not remove the figures from this section.

11. Remove FSAR pages 1.2-25 through 1.2-30, Proposed Rev. 2B and replace with enclosed FSAR pages 1.2-25 through 1.2-30, Proposed Rev. 2C. Do not remove the figures from this section.

12. Remove FSAR page 1.6-1/2, Proposed Rev. 2B and replace with enclosed FSAR page 1.6-1/2, Proposed Rev. 2C. Do not remove the figures from this section.

13. Remove FSAR pages 1.D-3 through 1.D-5, Proposed Rev. 2B and replace with enclosed FSAR pages 1.D-3 through 1.D-6, Proposed Rev. 2C.

14. Remove FSAR pages 2.0-1 through 2.0-10, Proposed Rev. 2B and replace with enclosed FSAR pages 2.0-1 through 2.0-10, Proposed Rev. 2C.

15. Remove FSAR page 2.0-19/20, Proposed Rev. 2B and replace with enclosed FSAR page 2.0-19/20, Proposed Rev. 2C.

16. Remove FSAR pages 2.0-27 through 2.0-30, Proposed Rev. 2B and replace with enclosed FSAR pages 2.0-27 through 2.0-30, Proposed Rev. 2C.

17. Remove FSAR pages 2.1-7 through 2.1-10, Proposed Rev. 2B and replace with enclosed FSAR pages 2.1-7 through 2.1-10, Proposed Rev. 2C. Do not remove the figures from this section.

18. Remove FSAR pages 2.1-33 through 2.1-52, Proposed Rev. 2B and replace with enclosed FSAR pages 2.1-34 through 2.1-52, Proposed Rev. 2C. Do not remove the figures from this section.

19. Remove FSAR pages 2.2-1 through 2.2-10, Proposed Rev. 2B and replace with enclosed FSAR pages 2.2-1 through 2.2-10, Proposed Rev. 2C.

20. Remove FSAR page 2.2-21/22, Proposed Rev. 2B and replace with enclosed FSAR page 2.2-21/22, Proposed Rev. 2C.

21. Remove FSAR pages 2.2-43 through 2.2-46, Proposed Rev. 2B and replace with enclosed FSAR pages 2.2-43 through 2.2-46, Proposed Rev. 2C.

22. Remove FSAR pages 2.3-1 through 2.3-6, Proposed Rev. 2B and replace with enclosed FSAR pages 2.3-1 through 2.3-7, Proposed Rev. 2C.

23. Remove FSAR pages 2.6-1 through 2.6-3, Proposed Rev. 2B and replace with enclosed FSAR pages 2.6-1 through 2.6-3, Proposed Rev. 2C.

24. Insert new Appendix 2.C, (3 pages including Figure 2.C.1), Proposed Rev. 2C after existing Appendix 2.B, Proposed Rev. 2B.

25. Remove FSAR pages 4.0-1 through 4.0-5, Proposed Rev. 2B and replace with enclosed FSAR pages 4.0-1 through 4.0-5, Proposed Rev. 2C. Do not remove the figures from this section.

26. Remove FSAR pages 4.1-1 through 4.1-6, Proposed Rev. 2B and replace with enclosed FSAR pages 4.1-1 through 4.1-6, Proposed Rev. 2C.

27. Remove FSAR pages 4.3-1 through 4.3-3, Proposed Rev. 2B and replace with enclosed FSAR pages 4.3-1 through 4.3-4, Proposed Rev. 2C. Do not remove the figures from this section.

28. Remove FSAR pages 4.4-1 through 4.4-50, Proposed Rev. 2B and replace with enclosed FSAR pages 4.4-1 through 4.4-50, Proposed Rev. 2C. Do not remove the figures from this section.

29. Remove FSAR pages 4.5-1 through 4.5-24 Proposed Rev. 2B and replace with enclosed FSAR pages 4.5-1 through 4.5-24, Proposed Rev. 2C. Do not remove the figures from this section.

30. Remove FSAR page 4.6-1/2, Proposed Rev. 2B and replace with enclosed FSAR page 4.6-1/2, Proposed Rev. 2C.

31. Remove FSAR pages 4.7-1 through 4.7-3, Proposed Rev. 2B and replace with enclosed FSAR pages 4.7-1 through 4.7-3, Proposed Rev. 2C.

32. Remove FSAR pages 4.B-1 through 4.B-10, Proposed Rev. 2B and replace with enclosed FSAR pages 4.B-1 through 4.B-13, Proposed Rev. 2C. Do not remove the figures from this section.

33. Remove FSAR pages 8.1-1 through 8.1-41, Proposed Rev. 2B and replace with enclosed FSAR pages 8.1-1 through 8.1-42, Proposed Rev. 2C.

34. Remove FSAR pages 8.3-1 through 8.1-10, Proposed Rev. 2B and replace with enclosed FSAR pages 8.3-1 through 8.3-11, Proposed Rev. 2C.

35. Remove FSAR pages 9.1-1 through 9.1-16, Proposed Rev. 2B and replace with enclosed FSAR pages 9.1-1 through 9.1-16, Proposed Rev. 2C.

36. Remove FSAR page 10.3-3/4, Proposed Rev. 2B and replace with enclosed FSAR page 10.3-3/4, Proposed Rev. 2C.

37. Remove FSAR pages 11.1-1 through 11.1-16, Proposed Rev. 2B and replace with enclosed FSAR pages 11.1-1 through 11.1-20, Proposed Rev. 2C.

38. Remove FSAR pages 11.2-1 through 11.2-51, Proposed Rev. 2B and replace with enclosed FSAR pages 11.2-1 through 11.2-53, Proposed Rev. 2C. Do not remove the figures from this section.

39. Insert new Figure 11.2.8 at the end of Chapter 11.

40. Remove FSAR pages 12.1-1 through 12.1-3, Proposed Rev. 2B and replace with enclosed FSAR pages 12.1-1 through 12.1-3, Proposed Rev. 2C.

41. Remove FSAR pages 12.2-1 through 12.2-9, Proposed Rev. 2B and replace with enclosed FSAR pages 12.2-1 through 12.2-9, Proposed Rev. 2C.

42. Remove FSAR Appendix 12.A-Bases TOC, Proposed Rev. 2B and replace with enclosed FSAR Appendix 12.A-Bases TOC, Proposed Rev. 2C.

43. Remove FSAR Appendix 12.A pages B 3.1.2-1 through B 3.1.2-7, Proposed Rev. 2B and replace with enclosed FSAR Appendix 12.A B 3.1.2-1 through B 3.1.2-7, Proposed Rev. 2C.

44. Insert new FSAR Appendix 12.A pages B 3.1.4-1 through B 3.1.4-5, Proposed Rev. 2C after existing Appendix 12.A page B 3.1.3-7, Proposed Rev. 2B.

LAR 1014-2, REVISION 2, SUPPLEMENT 2 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® neutron poison material to allow the use of an alternate, equivalent neutron poison material, METAMIC®, 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® is proposed as an alternative to BORAL®. Because of the absence of interconnected porosities, the time required to dry a METAMIC®-equipped MPC is expected to be less compared to an MPC containing the rolled cermet class of neutron absorbers such as BORAL®.

Justification for Proposed Changes

METAMIC® neutron poison material has been demonstrated to be equivalent to BORAL® in performing the design function of absorbing thermal neutrons. METAMIC® is also equivalent to BORAL® in its thermal, structural, and shielding performance. The dimensions and tolerances for the fabrication and installation of the METAMIC® neutron absorber panels are identical to the current BORAL® dimensions and tolerances. The weight percent of B_4C in METAMIC® is less than that for BORAL® given the same panel thickness, ^{10}B areal density, and ^{10}B loading penalty (25%).

METAMIC® has been considered in the criticality analyses in the same manner as BORAL® previously was considered, with one exception: only a 10% penalty on ^{10}B loading was considered for METAMIC® versus the previous licensing basis value of 25% for BORAL®. This change is appropriate because METAMIC® 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® with 90% credit for ^{10}B.

EPRI Report 1003137, "Qualification of METAMIC® for Spent-Fuel Storage Applications" provides the pertinent qualification tests data for this material.

Holtec International proprietary Report HI-2022871, "Use of METAMIC® in Fuel Pool Applications", includes a detailed discussion of the use of METAMIC® in wet storage applications, but also includes information germane to dry storage. Both of these reports support the conclusion that METAMIC® 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 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) contain 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 changes 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 IV 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 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 2[1]. 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. They also provide optional requirements (fuel cladding hoop stress calculations) for MPC containing all moderate burnup fuel (≤ 45,000 MWD/MTU) to all 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.

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

[1] 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 (≤ 45,000 MWD/MTU). This is expected to be consistent with the soon-to-be-published ISG-11, Revision 3.

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 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 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.% ^{235}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[2]. 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

[2] 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 5th and March 7th, 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, 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 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 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°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}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 "μ" is the coefficient of friction between the overpack and the ISFSI pad, and incipient tipping, where the value of "μ" 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 "μ" 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 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.

TABLE OF CONTENTS

1.0 USE AND APPLICATION 1.1-1
1,1 Definitions 1.1-1
1.2 Logical Connectors 1.2-1
1.3 Completion Times 1.3-1
1.4 Frequency 1.4-1

2.0 ..... 2.0-1

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY 3.0-1
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY 3.0-2

3.1 SFSC INTEGRITY 3.1.1-1
3.1.1 Multi-Purpose Canister (MPC) 3.1.1-1
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 ~~TRANSFER CASK Average Surface Dose Rates~~ *Deleted* 3.2.1-1
3.2.2 TRANSFER CASK Surface Contamination 3.2.2-1
3.2.3 ~~OVERPACK Average Surface Dose Rates~~ *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 ~~Model-Dependent Limits~~*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 ~~Fuel Cladding Oxide Thickness Evaluation Program~~ *Deleted* 5.0-5
5.7 Radiation Protection Program 5.0-6

Table 5-1 TRANSFER CASK and OVERPACK Lifting Requirements 5.0-4

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 (≤ 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

AND

c. The MPC decay heat load exceeds either of the following limits based on orientation:

i. > 23 kW in the vertical orientation

OR

ii. > 19 kW in the horizontal orientation

OR

d. The MPC contains fuel of any authorized burnup

AND

e. The MPC decay heat load is > 30 kW

AND

f. The TRANSFER CASK is in the horizontal orientation

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Supplemental Cooling System inoperable.	A.1 With the TRANSFER CASK in the vertical orientation, restore SFSC Supplemental Cooling System to operable status.	7 days
	OR	
	A.2.1 With the TRANSFER CASK in the horizontal orientation, rotate the TRANSFER CASK to the vertical orientation.	24 hours
	AND	
	A.2.2 Restore the Supplemental Cooling System to operable status.	7 days
B. Required Actions A.1 through A.2.2 and associated Completion Times 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

Table 3-1
MPC Cavity Drying Limits

Fuel Burnup (MWD/MTU)	MPC Heat Load (kW)	Method of Moisture Removal (Notes 1 and 2)	Other Requirements (Note 3)
All Assemblies ≤ 45,000	≤ 17	VDS or FHD	None
All Assemblies ≤ 45,000	> 17 and ≤ 18	VDS or FHD	If VDS is used, annulus water recirculation is required
All Assemblies ≤ 45,000	> 18 and ≤ 38	FHD	None
One or more assemblies > 45,000	≤ 38	FHD	None

Notes:

1. VDS means Vacuum Drying System. The acceptance criterion for VDS is MPC cavity pressure shall be ≤ 3 torr for ≥ 30 minutes.

2. FHD means Forced Helium Dehydration System. The acceptance criterion for the FHD System is gas temperature exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes or gas dew point exiting the MPC shall be ≤ 22.9°F for ≥ 30 minutes.

3. Annulus water recirculation means a sufficient flow of water through the annulus between the MPC and the HI-TRAC inner shell during moisture removal operations (i.e., beginning prior to 10 torr descending and ending when helium backfill operations commence) to ensure exit water temperature is ≤ 125°F.

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)

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.

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.*

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.*

2.0 APPROVED CONTENTS

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) 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 (≥ 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

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[1]. 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.

---

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

Table 2.1-1 (page 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	≤ 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:	≤ 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)

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 ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~

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	≤ 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:	≤ 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 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% ^{235}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 ≥ 8 years and an average burnup ≤ 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, 14x14E, and 15x15G	≤ 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	≤ 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 ~~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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	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 24 of 339)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

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

i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 500 Watts
ii. All Other Array/Classes	As specified in *Section 2.4*. ~~Tables 2.1-5 or 2.1-7.~~

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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*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	*≤ 500 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 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 ~~cell~~ *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 ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MTU/MTIHM).
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~

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 ≥ 8 years and an average burnup ≤ 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 3~~1~~3 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	≤ 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:	≤ 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 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:	~~≤ 4.0 wt% ^{235}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 ≥ 8 years and an average burnup ≤ 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	≤ 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	≤ 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 and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E*F* 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

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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*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	*≤ 500 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 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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*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	*≤ 500 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 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.

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)
MPC-24/24E/24EF	≤ 1.583
MPC-32/32F	≤ 1.1875
MPC-68/68FF	≤ 0.522

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

The allowable maximum decay heat per fuel storage location for ZR-clad fuel in regionalized fuel loading shall be calculated as follows. Fuel loading regions for each MPC model are shown in Figures 2.1-1 through 2.1-4. The number of fuel storage locations in each region and the maximum total decay heat per MPC model is provided in Table 2.4-2.

Table 2.4-2

Fuel Storage Regions and Maximum Decay Heat per MPC

MPC Model	Number of Fuel Storage Locations in Region 1 ($N_{Region\ 1}$)	Number of Fuel Storage Locations in Region 2 ($N_{Region\ 2}$)	Maximum Decay Heat per MPC, Q (kW)
MPC-24/24E/24EF	4	20	38
MPC-32/32F	12	20	38
MPC-68/68FF	32	36	35.5

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

2.4.2.1 Choose a value of X between 1 and 6, where X is the ratio of the maximum decay heat per fuel storage location permitted in Region 1 ($q_{Region\ 1}$) to the maximum decay heat per fuel storage location permitted in Region 2 ($q_{Region\ 2}$).

2.4.2.2 Calculate $q_{Region\ 2}$ using the following equation:

$$q_{Region\ 2} = (2 \times Q)/[(1 + X^{0.15}) \times (N_{Region\ 1} \times X + N_{Region\ 2})] \qquad \text{Equation. 2.4.1}$$

Where:

$q_{Region\ 2}$ = *Maximum allowable decay heat per fuel storage location in Region 2 (kW)*

Q = *Maximum allowable heat load for the MPC model from Table 2.4-2 (kW)*

X = *Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step 2.4.2.1*

$N_{Region\ 1}$ = *Number of fuel storage locations in Region 1 from Table 2.4-2*

$N_{Region\ 2}$ = *Number of fuel storage locations in Region 2 from Table 2.4-2*

2.4.2.3 Calculate $q_{Region\ 1}$ using the following equation:

$$q_{Region\ 1} = X \times q_{Region\ 2} \qquad \text{Equation 2.4.2}$$

Where:

$q_{Region\ 1}$ = *Maximum allowable decay heat per fuel storage location in Region 1 (kW)*

$q_{Region\ 2}$ = *Maximum allowable decay heat per fuel storage location in Region 2 calculated in Step 2.4.2.2 (kW)*

X = *Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step 2.4.2.1*

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*

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

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).

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

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

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

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

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

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

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

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.

3.0 DESIGN FEATURES

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.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-24

1. Flux trap size: $\geq$ 1.09 in.

2. ^{10}B loading in the ~~Boral~~ neutron absorbers: $\geq$ 0.0267 g/cm^2 *(Boral) or* $\geq$ *0.0223 g/cm^2 (METAMIC)*

3.2.2 MPC-68 and MPC-68FF

1. Fuel cell pitch: $\geq$ 6.43 in.

2. ^{10}B loading in the ~~Boral~~ neutron absorbers: $\geq$ 0.0372 g/cm^2 *(Boral) or* $\geq$ *0.0310 g/cm^2 (METAMIC)*

3.2.3 MPC-68F

1. Fuel cell pitch: $\geq$ 6.43 in.

2. ^{10}B loading in the Boral neutron absorbers: $\geq$ 0.01 g/cm^2

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}B loading in the ~~Boral~~ neutron absorbers: $\geq$ 0.0372 g/cm^2 *(Boral) or* $\geq$ *0.0310 g/cm^2 (METAMIC)*

3.2.5 MPC-32 *and MPC-32F*

1. Fuel cell pitch: $\geq$ 9.158 inches

2. ^{10}B loading in the ~~Boral~~ neutron absorbers: $\geq$ 0.0372 g/cm^2 *(Boral) or* $\geq$ *0.0310 g/cm^2 (METAMIC)*

DESIGN FEATURES

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 ~~Boral~~ neutron poison region of the MPC basket with water in the MPC.

3.2.7 The B_4C content in METAMIC shall be ≤ 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

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.

(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
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.*	*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.* *Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.*
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.

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
MPC basket supports and lift lugs	*NB-1130*	*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.* *NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.*	*The 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.*

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.
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.*	*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.* *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.*

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.*	*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.* *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).*
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.*	*The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-TRANSFER CASK) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.*
MPC 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. ~~The MPC vent and drain cover plate welds are leak tested.~~ 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.	The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be ~~hydrostatically~~*pressure* tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant ~~hydrostatic~~*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. The inspection ~~process~~ *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 ~~weld is confirmed by leakage testing and liquid penetrant examination~~ and the closure ring welds ~~is~~ *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.

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.

Table 3-1 (page 7 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 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 ~~large~~ 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 ~~a minimum~~ *large* safety factors ~~of > 12~~ during lifting. Likewise, ~~T~~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 > 6 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 "*"~~. 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. *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.*

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*	*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).* *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.*

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 μ 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 μ.* Unless demonstrated by appropriate testing that a higher *coefficient of friction* value ~~of μ~~ is appropriate for a specific ISFSI, the value ~~of μ~~ used shall be 0.53. Representative values of G_H and G_V combinations for μ *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 (μ = 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

3.4 Site-Specific Parameters and Analyses (continued)

b. 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 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

(continued)

3.4 Site-Specific Parameters and Analyses (continued)

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 ≤ 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 ≤ 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.

(continued)

DESIGN FEATURES

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° 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.

(continued)

DESIGN FEATURES

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* of 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)

DESIGN FEATURES

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.

(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 ≥ 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

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 (≤ 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 ≤ 60.3 psig (75 psia).

3.6.2.3 The hourly recirculation rate of helium shall be ≥ 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 demoisturizer outlet is *verified by measurement to remain* ≤ 21°F for a period of 30 minutes *or if the dew point of the gas exiting the MPC is verified by measurement to remain ≤ 22.9°F for ≥ 30 minutes*.

3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point ≤ 120°F.

3.6.2.6 The demoisturizing 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.

(continued)

DESIGN FEATURES

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.

TABLE OF CONTENTS

1.0 USE AND APPLICATION 1.1-1
1,1 Definitions 1.1-1
1.2 Logical Connectors 1.2-1
1.3 Completion Times 1.3-1
1.4 Frequency 1.4-1

2.0 ..... 2.0-1

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY 3.0-1
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY 3.0-2

3.1 SFSC INTEGRITY 3.1.1-1
3.1.1 Multi-Purpose Canister (MPC) 3.1.1-1
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

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 (≤ 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

AND

c. The MPC decay heat load exceeds either of the following limits based on orientation:

i. > 23 kW in the vertical orientation

OR

ii. > 19 kW in the horizontal orientation

OR

d. The MPC contains fuel of any authorized burnup

AND

e. The MPC decay heat load is > 30 kW

AND

f. The TRANSFER CASK is in the horizontal orientation

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Supplemental Cooling System inoperable.	A.1 With the TRANSFER CASK in the vertical orientation, restore SFSC Supplemental Cooling System to operable status.	7 days
	OR	
	A.2.1 With the TRANSFER CASK in the horizontal orientation, rotate the TRANSFER CASK to the vertical orientation.	24 hours
	AND	
	A.2.2 Restore the Supplemental Cooling System to operable status.	7 days
B. Required Actions A.1 through A.2.2 and associated Completion Times 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

Table 3-1
MPC Cavity Drying Limits

Fuel Burnup (MWD/MTU)	MPC Heat Load (kW)	Method of Moisture Removal (Notes 1 and 2)	Other Requirements (Note 3)
All Assemblies ≤ 45,000	≤ 17	VDS or FHD	None
All Assemblies ≤ 45,000	> 17 and ≤ 18	VDS or FHD	If VDS is used, annulus water recirculation is required
All Assemblies ≤ 45,000	> 18 and ≤ 38	FHD	None
One or more assemblies > 45,000	≤ 38	FHD	None

Notes:

1. VDS means Vacuum Drying System. The acceptance criterion for VDS is MPC cavity pressure shall be ≤ 3 torr for ≥ 30 minutes.

2. FHD means Forced Helium Dehydration System. The acceptance criterion for the FHD System is gas temperature exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes or gas dew point exiting the MPC shall be ≤ 22.9°F for ≥ 30 minutes .

3. Annulus water recirculation means a sufficient flow of water through the annulus between the MPC and the HI-TRAC inner shell during moisture removal operations (i.e., beginning prior to 10 torr descending and ending when helium backfill operations commence) to ensure exit water temperature is ≤ 125°F.

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.

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

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.

2.0 APPROVED CONTENTS

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. Deleted.

d. 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.

e. 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)

2.0 Approved Contents

2.1 Fuel Specifications and Loading Conditions (cont'd)

2.1.3 Regionalized Fuel Loading

Users may choose to store fuel using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Regionalized loading is limited to 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[1]. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Section 2.4.2. Fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3.

2.2 Violations

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

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

2.2.2 Within 24 hours, notify the NRC Operations Center.

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

2.3 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.

---

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

Table 2.1-1 (page 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 — ≤ 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)

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 ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Section 2.4.

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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	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:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 500 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 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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	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	≤ 500 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 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 ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MTU/MTIHM).
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Section 2.4.

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 ≥ 8 years and an average burnup ≤ 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 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 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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	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	≤ 500 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 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 ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	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	≤ 500 Watts.
ii. All Other Array/Classes	As specified in Section 2.3.
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 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.

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)
MPC-24/24E/24EF	≤ 1.583
MPC-32/32F	≤ 1.1875
MPC-68/68FF	≤ 0.522

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

The allowable maximum decay heat per fuel storage location for ZR-clad fuel in regionalized fuel loading shall be calculated as follows. Fuel loading regions for each MPC model are shown in Figures 2.1-1 through 2.1-4. The number of fuel storage locations in each region and the maximum total decay heat per MPC model is provided in Table 2.4-2.

Table 2.4-2

Fuel Storage Regions and Maximum Decay Heat per MPC

MPC Model	Number of Fuel Storage Locations in Region 1 ($N_{Region\ 1}$)	Number of Fuel Storage Locations in Region 2 ($N_{Region\ 2}$)	Maximum Decay Heat per MPC, Q (kW)
MPC-24/24E/24EF	4	20	38
MPC-32/32F	12	20	38
MPC-68/68FF	32	36	35.5

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

2.4.2.1 Choose a value of X between 1 and 6, where X is the ratio of the maximum decay heat per fuel storage location permitted in Region 1 ($q_{Region\ 1}$) to the maximum decay heat per fuel storage location permitted in Region 2 ($q_{Region\ 2}$).

2.4.2.2 Calculate $q_{Region\ 2}$ using the following equation:

$$q_{Region\ 2} = (2 \times Q)/[(1 + X^{0.15}) \times (N_{Region\ 1} \times X + N_{Region\ 2})] \qquad \text{Equation. 2.4.1}$$

Where:

$q_{Region\ 2}$ = Maximum allowable decay heat per fuel storage location in Region 2 (kW)

Q = Maximum allowable heat load for the MPC model from Table 2.4-2 (kW)

X = Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step 2.4.2.1

$N_{Region\ 1}$ = Number of fuel storage locations in Region 1 from Table 2.4-2

$N_{Region\ 2}$ = Number of fuel storage locations in Region 2 from Table 2.4-2

2.4.2.3 Calculate $q_{Region\ 1}$ using the following equation:

$$q_{Region\ 1} = X \times q_{Region\ 2} \qquad \text{Equation 2.4.2}$$

Where:

$q_{Region\ 1}$ = Maximum allowable decay heat per fuel storage location in Region 1 (kW)

$q_{Region\ 2}$ = Maximum allowable decay heat per fuel storage location in Region 2 calculated in Step 2.4.2.2 (kW)

X = Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step 2.4.2.1

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

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

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)

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

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

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

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

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

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

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

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.

3.0 DESIGN FEATURES

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.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-24

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

3.2.2 MPC-68 and MPC-68FF

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

3.2.3 MPC-68F

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

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}B loading in the neutron absorbers: $\geq$ 0.0372 g/cm^2 (Boral) and $\geq$ 0.0310 g/cm^2 (METAMIC)

3.2.5 MPC-32 and MPC-32F

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

DESIGN FEATURES

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)

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.

(continued)

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.	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. Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.

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	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. NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.	The 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.
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.	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. 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.

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.	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. 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).
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.	The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-TRANSFER CASK) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.
MPC 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 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.	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. 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.

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.	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. 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.

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	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). 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.

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 μ 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 μ. 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 (μ = 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)

3.4 Site-Specific Parameters and Analyses (continued)

b. 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 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

(continued)

3.4 Site-Specific Parameters and Analyses (continued)

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.

(continued)

DESIGN FEATURES

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 ≥ 0° 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.

(continued)

DESIGN FEATURES

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)

DESIGN FEATURES

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.

(continued)

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 ≥ 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

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 (≤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 ≤ 60.3 psig (75 psia).

3.6.2.3 The hourly recirculation rate of helium shall be ≥ 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 demoisturizer outlet is verified by measurement to remain ≤ 21°F for a period of 30 minutes or if the dew point of the gas exiting the MPC is verified by measurement to remain ≤ 22.9°F for ≥ 30 minutes.

3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point ≤ 120°F.

3.6.2.6 The demoisturizing 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.

(continued)

DESIGN FEATURES

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.

HI-STORM 100 FSAR
TABLE OF CONTENTS

CHAPTER 1: GENERAL DESCRIPTION 1.0-1

1.0 GENERAL INFORMATION 1.0-1
1.0.1 Engineering Change Orders 1.0-3

1.1 INTRODUCTION 1.1-1

1.2 GENERAL DESCRIPTION OF HI-STORM 100 SYSTEM 1.2-1
1.2.1 System Characteristics 1.2-1
1.2.1.1 Multi-Purpose Canisters 1.2-3
1.2.1.2 Overpacks 1.2-6
1.2.1.2.1 HI-STORM 100 Overpack (Storage) 1.2-6
1.2.1.2.2 HI-TRAC (Transfer Cask) – Standard Design 1.2-10
1.2.1.2.3 HI-TRAC 125D Transfer Cask 1.2-11
1.2.1.3 Shielding Materials 1.2-11
1.2.1.3.1 Fixed Neutron Absorbers 1.2-13
1.2.1.3.2 Neutron Shielding 1.2-18
1.2.1.3.3 Gamma Shielding Material 1.2-20
1.2.1.4 Lifting Devices 1.2-21
1.2.1.5 Design Life 1.2-21
1.2.2 Operational Characteristics 1.2-22
1.2.2.1 Design Features 1.2-22
1.2.2.2 Sequence of Operations 1.2-23
1.2.2.3 Identification of Subjects for Safety and Reliability Analysis 1.2-28
1.2.2.3.1 Criticality Prevention 1.2-28
1.2.2.3.2 Chemical Safety 1.2-28
1.2.2.3.3 Operation Shutdown Modes 1.2-29
1.2.2.3.4 Instrumentation 1.2-29
1.2.2.3.5 Maintenance Technique 1.2-29
1.2.3 Cask Contents 1.2-29

1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS 1.3-1

1.4 GENERIC CASK ARRAYS 1.4-1

1.5 GENERAL ARRANGEMENT DRAWINGS 1.5-1

1.6 REFERENCES 1.6-1

APPENDIX 1.A: ALLOY X DESCRIPTION
APPENDIX 1.B: HOLTITE™ MATERIAL DATA
APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA
APPENDIX 1.D: REQUIREMENTS ON HI-STORM 100 SHIELDING CONCRETE

CHAPTER 2: PRINCIPAL DESIGN CRITERIA 2.0-1

2.0 PRINCIPAL DESIGN CRITERIA 2.0-1
2.0.1 MPC Design Criteria 2.0-1
2.0.2 HI-STORM 100 Overpack Design Criteria 2.0-6
2.0.3 HI-TRAC Transfer Cask Design Criteria 2.0-9

2.0.4 Principal Design Criteria for the ISFSI Pad 2.0-11
2.0.4.1 Design and Construction Criteria 2.0-11
2.0.4.2 Applicable Codes 2.0-12
2.0.4.3 Limiting Design Parameters 2.0-15
2.0.4.4 Anchored Cask/ISFSI Interface 2.0-16

2.1 SPENT FUEL TO BE STORED 2.1-1
2.1.1 Determination of The Design Basis Fuel 2.1-1
2.1.2 Intact SNF Specifications 2.1-2
2.1.3 Damaged SNF and Fuel Debris Specifications 2.1-2
2.1.4 Deleted 2.1-3
2.1.5 Structural Parameters for Design Basis SNF 2.1-3
2.1.6 Thermal Parameters for Design Basis SNF 2.1-3
2.1.7 Radiological Parameters for Design Basis SNF 2.1-5
2.1.8 Criticality Parameters for Design Basis SNF 2.1-5
2.1.9 Summary of SNF Design Criteria 2.1-6

2.2 HI-STORM 100 DESIGN CRITERIA 2.2-1
2.2.1 Normal Condition Design Criteria 2.2-2
2.2.1.1 Dead Weight 2.2-2
2.2.1.2 Handling 2.2-2
2.2.1.3 Pressure 2.2-2
2.2.1.4 Environmental Temperatures 2.2-3
2.2.1.5 Design Temperatures 2.2-4
2.2.1.6 Snow and Ice 2.2-5
2.2.2 Off-Normal Conditions Design Criteria 2.2-5
2.2.2.1 Pressure 2.2-5
2.2.2.2 Environmental Temperatures 2.2-5
2.2.2.3 Design Temperatures 2.2-6
2.2.2.4 Leakage of One Seal 2.2-6
2.2.2.5 Partial Blockage of Air Inlets 2.2-6
2.2.2.6 Off-Normal HI-TRAC Handling 2.2-7
2.2.3 Environmental Phenomena and Accident Condition Design Criteria 2.2-7
2.2.3.1 Handling Accident 2.2-7
2.2.3.2 Tip-Over 2.2-9
2.2.3.3 Fire 2.2-10
2.2.3.4 Partial Blockage of MPC Basket Vent Holes 2.2-10
2.2.3.5 Tornado 2.2-11
2.2.3.6 Flood 2.2-11
2.2.3.7 Seismic Design Loadings 2.2-12
2.2.3.8 100% Fuel Rod Rupture 2.2-12
2.2.3.9 Confinement Boundary Leakage 2.2-13
2.2.3.10 Explosion 2.2-13
2.2.3.11 Lightning 2.2-13
2.2.3.12 Burial Under Debris 2.2-13
2.2.3.13 100% Blockage of Air Inlets 2.2-14
2.2.3.14 Extreme Environmental Temperature 2.2-14
2.2.3.15 Bounding Hydraulic, Wind, and Missile Loads for Anchored HI-STORMA2.2-14
2.2.4 Applicability of Governing Documents 2.2-14

2.2.5 Service Limits 2.2-16
2.2.6 Loads 2.2-16
2.2.7 Load Combinations 2.2-17
2.2.8 Allowable Stresses 2.2-18

2.3 SAFETY PROTECTION SYSTEMS 2.3-1
2.3.1 General 2.3-1
2.3.2 Protection by Multiple Confinement Barriers and Systems 2.3-2
2.3.2.1 Confinement Barriers and Systems 2.3-2
2.3.2.2 Cask Cooling 2.3-2
2.3.3 Protection by Equipment and Instrumentation Selection 2.3-3
2.3.3.1 Equipment 2.3-3
2.3.3.2 Instrumentation 2.3-16
2.3.4 Nuclear Criticality Safety 2.3-16
2.3.4.1 Control Methods for Prevention of Criticality 2.3-17
2.3.4.2 Error Contingency Criteria 2.3-17
2.3.4.3 Verification Analyses 2.3-17
2.3.5 Radiological Protection 2.3-17
2.3.5.1 Access Control 2.3-17
2.3.5.2 Shielding 2.3-18
2.3.5.3 Radiological Alarm System 2.3-19
2.3.6 Fire and Explosion Protection 2.3-19

2.4 DECOMMISSIONING CONSIDERATIONS 2.4-1

2.5 REGULATORY COMPLIANCE 2.5-1

2.6 REFERENCES 2.6-1

APPENDIX 2.A: GENERAL DESIGN AND CONSTRUCTION REQUIREMENTS FOR THE ISFSI PAD FOR HI-STORM 100A

APPENDIX 2.B: THE FORCED HELIUM DEHYDRATION (FHD) SYSTEM

APPENDIX 2.C: THE SUPPLEMENTAL COOLING SYSTEM

CHAPTER 3: STRUCTURAL EVALUATION 3.0-1

3.1 STRUCTURAL DESIGN 3.1-1
3.1.1 Discussion 3.1-1
3.1.2 Design Criteria 3.1-5
3.1.2.1 Loads and Load Combinations 3.1-8
3.1.2.1.1 Individual Load Cases 3.1-8
3.1.2.1.2 Load Combinations 3.1-13
3.1.2.2 Allowables 3.1-17
3.1.2.3 Brittle Fracture 3.1-19
3.1.2.4 Fatigue 3.1-21
3.1.2.5 Buckling 3.1-21

3.2 WEIGHTS AND CENTERS OF GRAVITY 3.2-1

3.3 MECHANICAL PROPERTIES OF MATERIALS 3.3-1
3.3.1 Structural Materials 3.3-1
3.3.1.1 Alloy X 3.3-1
3.3.1.2 Carbon Steel, Low-Alloy and Nickel Alloy Steel 3.3-2
3.3.1.3 Bolting Materials 3.3-2
3.3.1.4 Weld Material 3.3-2
3.3.2 Nonstructural Materials 3.3-3
3.3.2.1 Solid Neutron Shield 3.3-3
3.3.2.2 Solid Neutron Absorber 3.3-3
3.3.2.3 Concrete 3.3-3
3.3.2.4 Lead 3.3-3
3.3.2.5 Aluminum Heat Conduction Elements 3.3-3

3.4 GENERAL STANDARDS FOR CASKS 3.4-1
3.4.1 Chemical and Galvanic Reactions 3.4-1
3.4.2 Positive Closure 3.4-2
3.4.3 Lifting Devices 3.4-2
3.4.3.1 125 Ton HI-TRAC Lifting Analysis - Trunnions 3.4-4
3.4.3.2 125 Ton HI-TRAC Lifting - Trunnion Lifting Block Welds, Bearing, and Thread Shear Stress (Region A) 3.4-5
3.4.3.3 125 Ton HI-TRAC Lifting - Structure near Trunnion (Region B/Region A) 3.4-6
3.4.3.4 100 Ton HI-TRAC Lifting Analysis 3.4-6
3.4.3.5 HI-STORM 100 Lifting Analyses 3.4-7
3.4.3.6 MPC Lifting Analysis 3.4-11
3.4.3.7 Miscellaneous Lid Lifting Analyses 3.4-11
3.4.3.8 HI-TRAC Pool Lid Analysis - Lifting MPC From the Spent Fuel Pool (Load Case 01 in Table 3.1.5) 3.4-12
3.4.3.9 HI-TRAC Transfer Lid Analysis – Lifting MPC Away From Spent Fuel Pool (Load Case 01 in Table 3.1.5) 3.4-14
3.4.3.10 HI-TRAC Bottom Flange Evaluation during Lift (Load Case 01 in Table 3.1.5) 3.4-15
3.4.3.11 Conclusion 3.4-16
3.4.4 Heat 3.4-16
3.4.4.1 Summary of Pressures and Temperatures 3.4-16
3.4.4.2 Differential Thermal Expansion 3.4-16
3.4.4.2.1 Normal Hot Environment 3.4-16
3.4.4.2.2 Fire Accident 3.4-18
3.4.4.3 Stress Calculations 3.4-20
3.4.4.3.1 MPC Stress Calculations 3.4-21
3.4.4.3.2 HI-STORM 100 Storage Overpack Stress Calculations 3.4-37
3.4.4.3.3 HI-TRAC Transfer Cask Stress Calculations 3.4-45
3.4.4.4 Comparison with Allowable Stresses 3.4-54
3.4.4.4.1 MPC 3.4-54
3.4.4.4.2 Storage Overpack and HI-TRAC 3.4-56
3.4.4.5 Elastic Stability Considerations 3.4-56
3.4.4.5.1 MPC Elastic Stability 3.4-56

3.4.4.5.2 HI-STORM 100 Overpack Elastic Stability 3.4-56
3.4.5 Cold 3.4-57
3.4.6 HI-STORM 100 Kinematic Stability Under Flood Condition (Load Case A in Table 3.1.1) 3.4-59
3.4.7 Seismic Event and Explosion - HI-STORM 100 ..... 3.4-61
3.4.7.1 Seismic Event (Load Case C in Table 3.1.1) 3.4-62
3.4.7.2 Explosion (Load Case 05 in Table 3.1.5) 3.4-72
3.4.7.3 Anchored HI-STORM Systems Under High-Seismic DBE (Load Case C in Table 3.1.1) 3.4-75
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) 3.4-84
3.4.8.1 HI-STORM 100 Storage Overpack 3.4-85
3.4.8.2 HI-TRAC Transfer Cask 3.4-89
3.4.8.2.1 Intermediate Missile Strike 3.4-89
3.4.8.2.2 Large Missile Strike 3.4-92
3.4.9 HI-TRAC Drop Events (Load Case 02.b in Table 3.1.5) 3.4-94
3.4.9.1 Working Model 2D Analysis of Drop Event 3.4-94
3.4.9.2 DYNA3D Analysis of Drop Event 3.4-95
3.4.9.3 Horizontal Drop of HI-TRAC 125D 3.4-98
3.4.10 HI-STORM 100 Non-Mechanistic Tip-Over and Vertical Drop Event (Load Cases 02.a and 0.2c in Table 3.1.5) 3.4-99
3.4.11 Storage Overpack and HI-TRAC Transfer Cask Service Life 3.4-102
3.4.11.1 Storage Overpack 3.4-103
3.4.11.2 Transfer Cask 3.4-104
3.4.12 MPC Service Life 3.4-105
3.4.13 Design and Service Life 3.4-107

3.5 FUEL RODS 3.5-1

3.6 SUPPLEMENTAL DATA 3.6-1
3.6.1 Additional Codes and Standards Referenced in HI-STORM 100 System Design and Fabrication 3.6-1
3.6.2 Computer Programs 3.6-7
3.6.3 Appendices Included in Chapter 3 ..... 3.6-8
3.6.4 Calculation Package 3.6-9

3.7 COMPLIANCE WITH NUREG-1536 ..... 3.7-1

3.8 REFERENCES 3.8-1

APPENDIX 3.A HI-STORM DECELERATION UNDER POSTULATED VERTICAL DROP EVENT AND TIPOVER
APPENDIX 3.B DELETED
APPENDIX 3.C DELETED
APPENDIX 3.D DELETED
APPENDIX 3.E DELETED
APPENDIX 3.F DELETED
APPENDIX 3.G DELETED
APPENDIX 3.H DELETED

APPENDIX 3.I DELETED
APPENDIX 3.J DELETED
APPENDIX 3.K DELETED
APPENDIX 3.L DELETED
APPENDI X 3.M DELETED
APPENDIX 3.N DELETED
APPENDIX 3.O DELETED
APPENDIX 3.P DELETED
APPENDIX 3.Q DELETED
APPENDIX 3.R DELETED
APPENDIX 3.S DELETED
APPENDIX 3.T DELETED
APPENDIX 3.U DELETED
APPENDIX 3.V DELETED
APPENDIX 3.W DELETED
APPENDIX 3.X DELETED
APPENDIX 3.Y DELETED
APPENDIX 3.Z DELETED
APPENDIX 3.AA DELETED
APPENDIX 3.AB DELETED
APPENDIX 3.AC DELETED
APPENDIX 3.AD DELETED
APPENDIX 3.AE DELETED
APPENDIX 3.AF DELETED
APPENDIX 3.AG DELETED
APPENDIX 3.AH DELETED
APPENDIX 3.AI DELETED
APPENDIX 3.AJ DELETED
APPENDIX 3.AK DELETED
APPENDIX 3.AL DELETED
APPENDIX 3.AM DELETED
APPENDIX 3.AN DELETED
APPENDIX 3.AO DELETED
APPENDIX 3.AP DELETED
APPENDIX 3.AQ DELETED
APPENDIX 3.AR DELETED
APPENDIX 3.AS DELETED

CHAPTER 4: THERMAL EVALUATION 4.0-1

4.0 OVERVIEW 4.0-1

4.1 DISCUSSION 4.1-1

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS 4.2-1

4.3 SPECIFICATIONS FOR COMPONENTS 4.3-1
4.3.1 Deleted
4.3.1.1 Deleted

4.3.1.2 Deleted
4.3.2 Deleted
4.3.3 Deleted

4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE....4.4-1
4.4.1 Thermal Model....4.4-1
4.4.1.1 Analytical Model - General Remarks....4.4-2
4.4.1.1.1 Overview of the Thermal Model....4.4-3
4.4.1.1.2 Fuel Region Effective Thermal Conductivity Calculation....4.4-7
4.4.1.1.3 Effective Thermal Conductivity of Boral/ Sheathing/Box Wall Sandwich....4.4-9
4.4.1.1.4 Modeling of Basket Conductive Heat Transport....4.4-10
4.4.1.1.5 Deleted....4.4-12
4.4.1.1.6 Deleted....4.4-12
4.4.1.1.7 Annulus Air Flow and Heat Exchange....4.4-12
4.4.1.1.8 Determination of Solar Heat Input....4.4-13
4.4.1.1.9 FLUENT Model for HI-STORM....4.4-14
4.4.1.1.10 Deleted....4.4-18
4.4.1.1.11 Deleted....4.4-18
4.4.1.1.12 Deleted....4.4-18
4.4.1.1.13 Deleted....4.4-19
4.4.1.1.14 MPC Helium Backfill Pressure....4.4-19
4.4.1.2 Test Model....4.4-20
4.4.2 Maximum Temperatures....4.4-20
4.4.3 Minimum Temperatures....4.4-22
4.4.4 Maximum Internal Pressure....4.4-22
4.4.5 Maximum Thermal Stresses....4.4-23
4.4.6 Evaluation of System Performance for Normal Conditions of Storage....4.4-25

4.5 THERMAL EVALUATION OF SHORT TERM OPERATING CONDITIONS....4.5-1
4.5.1 Synopsis of Short Term Operating Conditions....4.5-1
4.5.1.1 Supplemental Cooling System....4.5-2
4.5.2 Thermal Acceptance Criteria for Short Term Operations....4.5-2
4.5.3 The HI-TRAC Thermal Model....4.5-4
4.5.3.1 Overview....4.5-3
4.5.3.2 Effective Thermal Conductivity of Water Jacket....4.5-5
4.5.3.3 Heat Rejection from Transfer Cask Exterior Surfaces....4.5-5
4.5.3.4 Determination of Solar Heat Input....4.5-6
4.5.3.5 Lead-to-Steel Interface....4.5-6
4.5.4 Loading Operations with Flooded MPC....4.5-6
4.5.5 MPC Drying....4.5-9
4.5.5.1 Drying Options....4.5-9
4.5.5.2 Forced Helium Dehydration....4.5-9
4.5.5.3 Vacuum Drying....4.5-9
4.5.6 Onsite Transport in HI-TRAC....4.5-11
4.5.6.1 Analysis....4.5-

11
4.5.6.2 Results....4.5-11
4.5.7 MPC Cooldown and Reflooding for Defueling Operations....4.5-13
4.5.8 Minimum Temperatures for On-Site Transport....4.5-14
4.5.9 Evaluation of System Performance for Normal Conditions of Handling and Onsite Transport....4.5-14

4.6 REGULATORY COMPLIANCE....4.6-1
4.6.1 Normal Conditions of Storage....4.6-1
4.6.2 Short Term Operations....4.6-2

4.7 REFERENCES....4.7-1

APPENDIX 4.A DELETED
APPENDIX 4.B CONSERVATISMS IN THE THERMAL ANALYSIS OF THE HI-STORM 100 SYSTEM

CHAPTER 5: SHIELDING EVALUATION....5.0-1

5.0 INTRODUCTION....5.0-1

5.1 DISCUSSION AND RESULTS....5.1-1
5.1.1 Normal and Off-Normal Operations....5.1-4
5.1.2 Accident Conditions....5.1-8

5.2 SOURCE SPECIFICATION....5.2-1
5.2.1 Gamma Source....5.2-2
5.2.2 Neutron Source....5.2-4
5.2.3 Stainless Steel Clad Fuel Source....5.2-6
5.2.4 Non-fuel Hardware....5.2-6
5.2.4.1 BPRAs and TPDs....5.2-7
5.2.4.2 CRAs and APSRs....5.2-9
5.2.5 Choice of Design Basis Assembly....5.2-10
5.2.5.1 PWR Design Basis Assembly....5.2-10
5.2.5.2 BWR Design Basis Assembly....5.2-11
5.2.5.3 Decay Heat Loads....5.2-13
5.2.6 Thoria Rod Canister....5.2-15
5.2.7 Fuel Assembly Neutron Sources....5.2-16
5.2.8 Stainless Steel Channels....5.2-16

5.3 MODEL SPECIFICATIONS....5.3-1
5.3.1 Description of the Radial and Axial Shielding Configuration....5.3-1
5.3.1.1 Fuel Configuration....5.3-6
5.3.1.2 Streaming Considerations....5.3-6
5.3.2 Regional Densities....5.3-7

5.4 SHIELDING EVALUATION....5.4-1

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations 5.4-4
5.4.2 Damaged Fuel Post-Accident Shielding Evaluation 5.4-6
5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel 5.4-6
5.4.2.2 Generic PWR and BWR Damaged Fuel 5.4-6
5.4.3 Site Boundary Evaluation 5.4-8
5.4.4 Stainless Steel Clad Fuel Evaluation 5.4-10
5.4.5 Mixed Oxide Fuel Evaluation 5.4-10
5.4.6 Non-Fuel Hardware 5.4-11
5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources 5.4-12
5.4.8 Thoria Rod Canister 5.4-13
5.4.9 Regionalized Dose Rate Evaluation 5.4-13

5.5 REGULATORY COMPLIANCE 5.5-1

5.6 REFERENCES 5.6-1

APPENDIX 5.A SAMPLE INPUT FILE FOR SAS2H
APPENDIX 5.B SAMPLE INPUT FILE FOR ORIGEN-S
APPENDIX 5.C SAMPLE INPUT FILE FOR MCNP
APPENDIX 5.D DOSE RATE COMPARISON FOR DIFFERENT COBALT IMPURITY LEVELS
APPENDIX 5.E DOSE RATES FOR A HI-STORM 100 OVERPACK WITH AND WITHOUT AN INNER SHIELD SHELL

CHAPTER 6: CRITICALITY EVALUATION 6.1-1

6.1 DISCUSSION AND RESULTS 6.1-2

6.2 SPENT FUEL LOADING 6.2-1
6.2.1 Definition of Assembly Classes 6.2-1
6.2.2 Intact PWR Fuel Assemblies 6.2-2
6.2.2.1 Intact PWR Fuel Assemblies in the MPC-24 without Soluble Boron 6.2-2
6.2.2.2 Intact PWR Assemblies in the MPC-24 with Soluble Boron 6.2-3
6.2.2.3 Intact PWR Assemblies in the MPC-24E and MPC-24EF with and without Soluble Boron 6.2-4
6.2.2.4 Intact PWR Assemblies in the MPC-32 ..... 6.2-4
6.2.3 Intact BWR Fuel Assemblies in the MPC-68 and MPC-68FF 6.2-5
6.2.4 BWR and PWR Damaged Fuel Assemblies and Fuel Debris 6.2-6
6.2.4.1 Damaged BWR Fuel Assemblies and BWR Fuel Debris in Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A 6.2-7
6.2.4.2 Damaged BWR Fuel Assemblies and Fuel Debris in the MPC-68 and MPC-68FF 6.2-7
6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-24E and MPC-24EF 6.2-8
6.2.5 Thoria Rod Canister 6.2-9

6.3 MODEL SPECIFICATION 6.3-1
6.3.1 Description of Calculational Model 6.3-1

6.3.2 Cask Regional Densities 6.3-3

6.4 CRITICALITY CALCULATIONS 6.4-1
6.4.1 Calculational or Experimental Method 6.4-1
6.4.1.1 Basic Criticality Safety Calculations 6.4-1
6.4.2 Fuel Loading or Other Contents Loading Optimization 6.4-2
6.4.2.1 Internal and External Moderation 6.4-2
6.4.2.1.1 Unborated Water 6.4-2
6.4.2.1.2 Borated Water 6.4-3
6.4.2.2 Partial Flooding 6.4-4
6.4.2.3 Clad Gap Flooding 6.4-4
6.4.2.4 Preferential Flooding 6.4-5
6.4.2.5 Design Basis Accidents 6.4-6
6.4.3 Criticality Results 6.4-6
6.4.4 Damaged Fuel and Fuel Debris 6.4-7
6.4.4.1 MPC-68, MPC-68f or MPC-68FF loaded with Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A 6.4-8
6.4.4.2 Generic BWR and PWR Damaged Fuel and Fuel Debris 6.4-9
6.4.4.2.1 Bounding Intact Assemblies 6.4-10
6.4.4.2.2 Bare Fuel Road Arrays 6.4-11
6.4.4.2.3 Distributed Enrichment in BWR Fuel 6.4-12
6.4.4.2.4 Results for MPC-68 and MPC-68FF 6.4-13
6.4.4.2.5 Results for MPC-24E and MPC-24EF 6.4-14
6.4.4.2.6 Results for MPC-32 and MPC-32F 6.4-14
6.4.5 Fuel Assemblies with Missing Rods 6.4-15
6.4.6 Thoria Rod Canister 6.4-16
6.4.7 Sealed Rods replacing BWR Water Rods 6.4-16
6.4.8 Non-Fuel Hardware in PWR Fuel Assemblies 6.4-16
6.4.9 Neutron Sources in Fuel Assemblies 6.4-17
6.4.10 Applicability of HI-STAR Analyses to HI-STORM 100 System 6.4-17
6.4.11 Fixed Neutron Absorber Matereial 6.4-18

6.5 CRITICALITY BENCHMARK EXPERIMENTS 6.5-1

6.6 REGULATORY COMPLIANCE 6.6-1

6.7 REFERENCES 6.7-1

APPENDIX 6.A BENCHMARK CALCULATIONS
APPENDIX 6.B DISTRIBUTED ENRICHMENTS IN BWR FUEL
APPENDIX 6.C CALCULATIONAL SUMMARY
APPENDIX 6.D SAMPLE INPUT FILES

CHAPTER 7: CONFINEMENT 7.0-1

7.0 INTRODUCTION 7.0-1

7.1 CONFINEMENT BOUNDARY 7.1-1
7.1.1 Confinement Vessel 7.1-1

7.1.2 Confinement Penetrations 7.1-2
7.1.3 Seals and Welds 7.1-3
7.1.4 Closure 7.1-3
7.1.5 Damaged Fuel Container 7.1-4
7.1.6 Design and Qualification of Final MPC Closure Welds 7.1-4

7.2 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE 7.2-1
7.2.1 Deleted 7.2-1
7.2.2 Deleted 7.2-1
7.2.3 Deleted 7.2-2
7.2.4 Deleted 7.2-2
7.2.5 Deleted 7.2-3
7.2.6 Deleted 7.2-3
7.2.7 Deleted 7.2-4
7.2.7.1 Deleted 7.2-4
7.2.7.2 Deleted 7.2-4
7.2.7.3 Deleted 7.2-6
7.2.7.4 Deleted 7.2-6
7.2.7.5 Deleted 7.2-6
7.2.7.6 Deleted 7.2-7
7.2.7.7 Deleted 7.2-7
7.2.7.8 Deleted 7.2-7
7.2.7.9 Deleted 7.2-7
7.2.8 Deleted 7.2-7
7.2.8.1 Deleted 7.2-8
7.2.8.2 Deleted 7.2-8
7.2.8.3 Deleted 7.2-8
7.2.9 Deleted 7.2-8

7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS 7.3-1
7.3.1 Deleted 7.3-1
7.3.2 Deleted 7.3-2
7.3.3 Deleted 7.3-3
7.3.3.1 Deleted 7.3-3
7.3.3.2 Deleted 7.3-5
7.3.3.3 Deleted 7.3-5
7.3.3.4 Deleted 7.3-5
7.3.3.5 Deleted 7.3-5
7.3.3.6 Deleted 7.3-6
7.3.3.7 Deleted 7.3-7
7.3.3.8 Deleted 7.3-7
7.3.3.9 Deleted 7.3-7
7.3.4 Deleted 7.3-7
7.3.4.1 Deleted 7.3-8
7.3.4.2 Deleted 7.3-8
7.3.5 Deleted 7.3-9
7.3.6 Deleted 7.3-9

7.4 REFERENCES 7.4-1

APPENDIX 7.A DELETED

CHAPTER 8: OPERATING PROCEDURES 8.0-1

8.0 INTRODUCTION 8.0-1

8.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL 8.1-1
- 8.1.1 Overview of Loading Operations 8.1-1
- 8.1.2 HI-TRAC and HI-STORM Receiving and Handling Operations 8.1-4
- 8.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation 8.1-7
- 8.1.4 MPC Fuel Loading 8.1-11
- 8.1.5 MPC Closure 8.1-11
- 8.1.6 Preparation for Storage 8.1-23
- 8.1.7 Placement of HI-STORM into Storage 8.1-25

8.2 ISFSI OPERATIONS 8.2-1

8.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL 8.3-1
- 8.3.1 Overview of HI-STORM 100 System Unloading Operations 8.3-1
- 8.3.2 HI-STORM Recovery From Storage 8.3-2
- 8.3.3 Preparation for Unloading 8.3-4
- 8.3.4 MPC Unloading 8.3-9
- 8.3.5 Post-Unloading Operations 8.3-9

8.4 MPC TRANSFER TO HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE 8.4-1
- 8.4.1 Overview of Operations 8.4-1
- 8.4.2 Recovery from Storage 8.4-1
- 8.4.3 MPC Transfer into the HI-STAR 100 Overpack 8.4-1

8.5 MPC TRANSFER TO HI-STORM DIRECTLY FROM TRANSPORT 8.5-1
- 8.5.1 Overview of Operations 8.5-1
- 8.5.2 HI-STAR Receipt and Preparation for MPC Transfer 8.5-2
- 8.5.3 Perform MPC Transfer into HI-STORM 100 ..... 8.5-4

8.6 REFERENCES 8.6-1

CHAPTER 9: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM 9.0-1

9.0 INTRODUCTION 9.0-1

9.1 ACCEPTANCE CRITERIA 9.1-1
- 9.1.1 Fabrication and Nondestructive Examination (NDE) 9.1-1
  - 9.1.1.1 MPC Lid-to-Shell Weld Volumetric Inspection 9.1-4
- 9.1.2 Structural and Pressure Tests 9.1-5
  - 9.1.2.1 Lifting Trunnions 9.1-5
  - 9.1.2.2 Pressure Testing 9.1-6
    - 9.1.2.2.1 HI-TRAC Transfer Cask Water Jacket 9.1-6
    - 9.1.2.2.2 MPC Confinement Boundary 9.1-7

9.1.2.3 Materials Testing 9.1-8
9.1.3 Leakage Testing 9.1-8
9.1.4 Component Tests 9.1-9
9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices 9.1-9
9.1.4.2 Seals and Gaskets 9.1-9

9.1.5 Shielding Integrity 9.1-9
9.1.5.1 Fabrication Testing and Control 9.1-10
9.1.5.2 Shielding Effectiveness Tests 9.1-11
9.1.5.3 Neutron Absorber Tests 9.1-12
9.1.6 Thermal Acceptance Tests 9.1-14
9.1.7 Cask Identification 9.1-15

9.2 MAINTENANCE PROGRAM 9.2-1
9.2.1 Structural and Pressure Parts 9.2-1
9.2.2 Leakage Tests 9.2-1
9.2.3 Subsystem Maintenance 9.2-2
9.2.4 Pressure Relief Valve 9.2-2
9.2.5 Shielding 9.2-2
9.2.6 Thermal 9.2-3

9.3 REGULATORY COMPLIANCE 9.3-1

9.4 REFERENCES 9.4-1

CHAPTER 10: RADIATION PROTECTION 10.1-1

10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA) 10.1-1
10.1.1 Policy Considerations 10.1-1
10.1.2 Design Considerations 10.1-2
10.1.3 Operational Considerations 10.1-4
10.1.4 Auxiliary/Temporary Shielding 10.1-5

10.2 RADIATION PROTECTION DESIGN FEATURES 10.2-1

10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT 10.3-1
10.3.1 Estimated Exposures for Loading and Unloading Operations 10.3-2
10.3.2 Estimated Exposures for Surveillance and Maintenance 10.3-3

10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT 10.4-1
10.4.1 Controlled Area Boundary Dose for Normal Operations 10.4-1
10.4.2 Controlled Area Boundary Dose for Off-Normal Conditions 10.4-2
10.4.3 Controlled Area Boundary Dose for Accident Conditions 10.4.2

10.5 REFERENCES 10.5-1

CHAPTER 11: ACCIDENT ANALYSIS 11.1-1

11.1 OFF-NORMAL CONDITIONS 11.1-1
11.1.1 Off-Normal Pressures 11.1-2
11.1.2 Off-Normal Environmental Temperatures 11.1-4
11.1.3 Leakage of One Seal 11.1-7
11.1.4 Partial Blockage of Air Inlets 11.1-9
11.1.5 Off-Normal Handling of HI-TRAC 11.1-12
11.1.6 Off-Normal Load Combinations 11.1-14

11.2 ACCIDENTS 11.2-1
11.2.1 HI-TRAC Transfer Cask Handling Accident 11.2-1
11.2.2 HI-STORM Overpack Handling Accident 11.2-4
11.2.3 Tip-Over 11.2-6
11.2.4 Fire Accident 11.2-8
11.2.5 Partial Blockage of MPC Basket Vent Holes 11.2-18
11.2.6 Tornado 11.2-20
11.2.7 Flood 11.2-22
11.2.8 Earthquake 11.2-24
11.2.9 100% Fuel Rod Rupture 11.2-25
11.2.10 Confinement Boundary Leakage 11.2-27
11.2.11 Explosion 11.2-30
11.2.12 Lightning 11.2-31
11.2.13 100% Blockage of Air Inlets 11.2-32
11.2.14 Burial Under Debris 11.2-37
11.2.15 Extreme Environmental Temperature 11.2-40

11.3 REFERENCES 11.3-1

CHAPTER 12: OPERATING CONTROLS AND LIMITS 12.0-1

12.0 INTRODUCTION 12.0-1

12.1 PROPOSED OPERATING CONTROLS AND LIMITS 12.1-1
12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria 12.1-1

12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS 12.2-1
12.2.1 Training Modules 12.2-1
12.2.2 Dry Run Training 12.2-2
12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings 12.2-3
12.2.4 Limiting Conditions for Operation 12.2-3
12.2.5 Equipment 12.2-3
12.2.6 Surveillance Requirements 12.2-3
12.2.7 Design Features 12.2-4
12.2.8 MPC 12.2-4
12.2.9 HI-STORM 100 Overpack 12.2-4
12.2.10 Decay Heat and Burnup Limits for Fuel Storage 12.2-4

12.3 TECHNICAL SPECIFICATIONS 12.3-1

12.4 REGULATORY EVALUATION 12.4-1

12.5 REFERENCES 12.5-1

APPENDIX 12.A TECHNICAL SPECIFICATION BASES FOR THE HOLTEC HI-STORM 100 SPENT FUEL STORAGE CASK SYSTEM
APPENDIX 12.B COMMENT RESOLUTION LETTERS

CHAPTER 13: QUALITY ASSURANCE 13.0-1

13.0 QUALITY ASSURANCE PROGRAM 13.0-1
13.0.1 Overview 13.0-1
13.0.2 Graded Approach to Quality Assurance 13.0-2

13.1 DELETED

13.2 DELETED

13.3 DELETED

13.4 DELETED

13.5 DELETED

13.6 REFERENCES 13.6-1

APPENDIX 13.A DELETED
APPENDIX 13.B DELETED

LIST OF FIGURES

1.1.1	HI-STORM 100 Overpack with MPC Partially Inserted
1.1.1A	HI-STORM 100S Overpack with MPC Partially Inserted
1.1.2	Cross Section Elevation View of MPC
1.1.3	HI-STORM 100 Overpack Cross Sectional Elevation View
1.1.3A	HI-STORM 100S Overpack Cross Sectional Elevation View
1.1.4	A Pictorial View of the HI-STORM 100A Overpack (100SA Model Shown)
1.1.5	Anchoring Detail for the HI-STORM 100A and 100SA Overpacks
1.2.1	Cross Section View of the HI-STORM 100 System
1.2.1A	Cross Section View of the HI-STORM 100S System
1.2.2	MPC-68 Cross Section View
1.2.3	MPC-32/32F Cross-Section
1.2.4	MPC-24/24E/24EF Cross Section View
1.2.5	Cross Section Elevation View of MPC
1.2.6	MPC Confinement Boundary
1.2.7	Cross Section of HI-STORM Overpack
1.2.8	HI-STORM 100 Overpack Cross Sectional Elevation View
1.2.8A	HI-STORM 100S Overpack Cross Sectional Elevation View
1.2.9	125-Ton HI-TRAC Transfer Cask with Pool Lid Cross Sectional Elevation View
1.2.9A	HI-TRAC 125D Transfer Cask Cross Sectional Elevation View
1.2.10	125-Ton HI-TRAC Transfer Cask with Transfer Lid Cross Sectional Elevation View
1.2.11	100-Ton HI-TRAC Transfer Cask with Pool Lid Cross Sectional Elevation View
1.2.12	100-Ton HI-TRAC Transfer Cask with Transfer Lid Cross Sectional Elevation View
1.2.13	DELETED
1.2.14	DELETED

1.2.15	DELETED
1.2.16a	Major HI-STORM 100 Loading Operations (Sheet 1 of 6)
1.2.16b	Major HI-STORM 100 Loading Operations (Sheet 2 of 6)
1.2.16c	Major HI-STORM 100 Loading Operations (Sheet 3 of 6)
1.2.16d	Major HI-STORM 100 Loading Operations (Sheet 4 of 6)
1.2.16e	Example of HI-STORM 100 Handling Options (Sheet 5 of 6)
1.2.16f	Example of HI-TRAC Handling Options (Sheet 6 of 6)
1.2.17a	Major HI-STORM 100 Unloading Operations (Sheet 1 of 4)
1.2.17b	Major HI-STORM 100 Unloading Operations (Sheet 2 of 4)
1.2.17c	Major HI-STORM 100 Unloading Operations (Sheet 3 of 4)
1.2.17d	Major HI-STORM 100 Unloading Operations (Sheet 4 of 4)
1.2.18	HI-STORM Mating Device
1.4.1	Cask Layout Pitch Requirements Based on 2 by N Array(s)
1.4.2	Cask Layout Pitch Requirements Based on a Square Array
1.A.1	Design Stress Intensity vs. Temperature
1.A.2	Tensile Strength vs. Temperature
1.A.3	Yield Stress vs. Temperature
1.A.4	Coefficient of Thermal Expansion vs. Temperature
1.A.5	Thermal Conductivity vs. Temperature
2.1.1	Damaged Fuel Container for Dresden Unit-1/Humboldt Bay SNF
2.1.2	TN Damaged Fuel Canister for Dresden Unit 1
2.1.2A	TN Thoria Rod Canister for Dresden Unit 1
2.1.2B	Holtec Damaged Fuel Container for PWR SNF in MPC-24E/24EF
2.1.2C	Holtec Damaged Fuel Container for BWR SNF in MPC-68/68FF
2.1.2D	Holtec Damaged Fuel Container for PWR SNF in MPC-32/32F

2.1.3 PWR Axial Burnup Profile with Normalized Distribution

2.1.4 BWR Axial Burnup Profile with Normalized Distribution

2.1.5 MPC With Upper and Lower Fuel Spacers

2.1.6 DELETED

2.1.7 DELETED

2.1.8 DELETED

2.1.9 Fuel Debris MPC ("F" Model)

2.3.1 HI-STAR Upending and Downending on a Rail Car

2.3.2 HI-TRAC Upending and Downending on a Heavy-Haul Transport Trailer

2.3.3 HI-TRAC Placement on HI-STORM 100 for MPC Transfer Operations

2.3.4 HI-TRAC Placement on HI-STAR 100 for MPC Transfer Operations

2.A.1 Typical HI-STORM/ISFSI Pad Fastening Detail

2.B.1 Schematic of the Forced Helium Dehydration System

2.C.1 Supplemental Cooling System Example P&I Diagram

3.1.1 MPC-68 and MPC-32 Fuel Basket Geometry

3.1.2 0° Drop Orientations for the MPCs

3.1.3 45° Drop Orientations for the MPCs

3.4.1 Finite Element Model of MPC-24 (0 Degree Drop Model)

3.4.2 Finite Element Model of MPC-32 (0 Degree Drop Model)

3.4.3 Finite Element Model of MPC-68 (0 Degree Drop Model)

3.4.4 Finite Element Model of MPC-24 (45 Degree Drop Model)

3.4.5 Finite Element Model of MPC-32 (45 Degree Drop Model)

3.4.6 Finite Element Model of MPC-68 (45 Degree Drop Model)

3.4.7 Detail of Fuel Assembly Pressure Load on MPC Basket

3.4.8 0 Degree Side Drop of MPC

3.4.9 45 Degree Side Drop of MPC

3.4.10 Comparison of 125 Ton and 100 Ton HI-TRAC Lifting Trunnion Connection

3.4.11 Confinement Boundary Model Showing Temperature Data Points

3.4.12 MPC-Confinement Boundary Finite Element Grid (Exploded View)

3.4.13 Von Mises Stress - Outer Shell

3.4.14 Plastic Strain Outer Shell

3.4.15 Von Mises Stress - Inner Shell

3.4.16 Plastic Strain - Inner Shell

3.4.16a Von Mises Stress - Channel

3.4.16b Plastic Strain - Channel

3.4.17 Top and Bottom Lifting of the Loaded HI-STORM 100

3.4.18 HI-TRAC Upending in the Upending Frame

3.4.19 HI-STORM 100 Tip-Over Event

3.4.20 HI-STORM 100 End Drop Event

3.4.21 HI-TRAC Lifting with the Pool and Transfer Lids

3.4.22 HI-TRAC Side Drop Event

3.4.23 Forces and Moments on 125 Ton Rotation Trunnion Weld

3.4.24 Working Model Solution for Impact Force on HI-TRAC 100 Transfer Cask Outer Shell

3.4.25 HI-STORM 100 Overturning Scenario - Initial Angular Velocity = 0.628 Radians/Second Assumed Caused By a Pressure Pulse

3.4.26 HI-STORM 100 Overturning Scenario - Initial Angular Velocity = 0.628 Radians/Second Maximum Angular Excursion

3.4.27 HI-TRAC Transfer Cask in Short-Side Impact (Cask Rests at a Position of -5° from Horizontal)

3.4.28 HI-TRAC Transfer Cask in Long-Side Impact (Cask Rests at a Position of -1° from Horizontal)

3.4.29 Free-Body of Transfer Lid During Primary Impact with Target

3.4.30 Seismic Spectra Sets Used for Time History Analysis of HI-STORM 100 on ISFSI Pad

3.4.31 RG 1.60 "H1"

3.4.32 RG 1.60 "H2"

3.4.33 RG 1.60 "VT"

3.4.34 Horizontal Acceleration Time History "FN"

3.4.35 Horizontal Acceleration Time History "FP"

3.4.36 Horizontal Acceleration Time History "FV"

3.4.37 Geometry for Quasi-Static Analysis

3.4.38 Free Body for Quasi-Static Analysis

3.4.39 Sector Lug Finite Element Mesh

3.4.40 Sector Lug Stress - Case 1 Preload

3.4.41 Sector Lug Stress Intensity - Case 2 Preload + Seismic

3.4.42 Exploded View Showing Ground Plane, Overpack, MPC, and Overpack Top Lid

3.4.43 View of Assembled HI-STORM on Pad-MPC Inside and Top Lid Attached

3.4.44 Variation of Foundation Resistance Force vs. Time for Reg. Guide 1.60 Seismic Input

3.4.45 Variation of Representative Stud Tensile Force vs. Time for Reg. Guide 1.60 Seismic Input

3.4.46 MPC/HI-STORM 100A Impulse vs. Time - Reg. Guide 1.60 Event

3.4.47 Instantaneous Calculated Coefficient of Friction - Reg. Guide 1.60 Event

3.4.48 HI-TRAC 125 Benchmark Simulation of Drop Scenario A

3.4.49 Simulation of HI-TRAC 125D 42" Horizontal Drop with Primary Impact at Top End Radial Support Tab

3.5.1 Fuel Rod Deformation Phases, $g_1 > 0$

3.5.2 Fuel Rod Deformation Phases, $g_1 = 0$

3.5.3 Fuel Rod Deformation Phases, $g_1 = 0$, $F_2 > F_1$

3.5.4 Inter-Grid Strap Deformation $F_3 > F_2$

3.5.5 Point Contact at Load F_4 Maximum Bending Moment at A

3.5.6 Extended Region of Contact $F_5 > F_4$, Zero Bending Moment at A'

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

3.5.8 View C - C

3.5.9 Exaggerated Detail Showing Multiple Fuel Rods Subject to Lateral Deflection with Final Stacking of Rod Column

3.A.1 Tipover Finite Element Model (3-D View)

3.A.2 Tipover Finite-Element Model (Plan)

3.A.3 Tipover Finite-Element Model (XZ View)

3.A.4 Tipover Finite-Element Model (YZ View)

3.A.5 End-Drop Finite-Element Model (3-D View)

3.A.6 End-Drop Finite-Element Model (Plan)

3.A.7 End-Drop Finite-Element Model (XZ View)

3.A.8 End-Drop Finite-Element Model (YZ View)

3.A.9 Soil Finite-Element Model (3-D View)

3.A.10 Concrete Pad Finite-Element Model (3-D View)

3.A.11 Overpack Steel Structure Finite-Element Model (3-D View)

3.A.12 Inner Shell and Channels Finite-Element Model (3-D View)

3.A.13 Lid Steel Finite-Element Model (3-D View)

3.A.14 Overpack Concrete Components Finite-Element Model (3-D View)

3.A.15 MPC Finite Element Model (3-D View)

3.A.16 Pivot Point During Tip-Over Condition

3.A.17 Tip-Over Event Overpack Slams Against the Foundation Developing a Resistive Force

3.A.18 Measurement Points and Corresponding Finite-Element Model Nodes

3.A.19 Deleted

3.A.20 Deleted

3.A.21 Deleted

3.A.22 Deleted

3.A.23 Deleted

3.A.24 Deleted

3.A.25 Deleted

3.A.26 Deleted

3.A.27 Deleted

3.A.28 Deleted

3.A.29 Deleted

3.A.30 Deleted

3.C.1 Deleted

3.C.2 Deleted

3.C.3 Deleted

3.D.1 Deleted

3.D.2a Deleted

3.D.2b Deleted

3.D.2c Deleted

3.D.3 Deleted

3.D.4a Deleted

3.D.4b Deleted

3.D.4c Deleted

3.D.5a Deleted

3.D.5b Deleted

3.D.5c Deleted

3.E.1 Deleted

3.E.2 Deleted

3.E.3 Deleted

3.F.1 Deleted

3.F.2 Deleted

3.F.3 Deleted

3.F.4 Deleted

3.G.1 Deleted

3.G.2 Deleted

3.G.3 Deleted

3.G.4 Deleted

3.G.5 Deleted

3.H.1 Deleted

3.I.1 Deleted

3.M.1 Deleted

3.U.1 Deleted

3.V.1 Deleted

3.W.1 Deleted

3.X.1 Deleted

3.X.2 Deleted

3.X.3 Deleted

3.X.4 Deleted

3.X.5 Deleted

3.Y.1 Deleted

3.Y.2 Deleted

3.Z.1 Deleted

3.Z.2 Deleted

3.Z.3 Deleted

3.Z.4 Deleted

3.Z.5 Deleted

3.Z.6 Deleted

3.AA.1 Deleted

3.AA.2 Deleted

3.AA.3 Deleted

3.AA.4 Deleted

3.AA.5 Deleted

3.AA.6 Deleted

3.AA.7 Deleted

3.AA.8 Deleted

3.AD.1 Deleted

3.AD.2 Deleted

3.AD.3 Deleted

3.AE.1 Deleted

3.AE.1a Deleted

3.AE.1b Deleted

3.AE.1c Deleted

3.AE.2 Deleted

3.AE.3 Deleted

3.AE.4 Deleted

3.AI.1 Deleted

3.AI.2 Deleted

3.AI.3 Deleted

3.AI.4 Deleted

3.AI.5 Deleted

3.AI.6 Deleted

3.AJ.1 Deleted

3.AJ.2 Deleted

3.AJ.3 Deleted

3.AN.1 Deleted

3.AN.2 Deleted

3.AN.3 Deleted

3.AN.4 Deleted

3.AN.5 Deleted

3.AN.6 Deleted

3.AN.7 Deleted

3.AN.8 Deleted

3.AN.9 Deleted

3.AN.10 Deleted

3.AN.11 Deleted

3.AN.12 Deleted

3.AN.13 Deleted

3.AN.14 Deleted

3.AN.15 Deleted

3.AN.16 Deleted

3.AN.17 Deleted

3.AN.18 Deleted

3.AN.19 Deleted

3.AN.20 Deleted

3.AN.21 Deleted

3.AN.22 Deleted

3.AN.23 Deleted

3.AN.24 Deleted

3.AN.25 Deleted

3.AN.26 Deleted

3.AN.27 Deleted

3.AN.28 Deleted

3.AN.29 Deleted

3.AN.30 Deleted

4.0.1 MPC Internal Helium Circulation

4.0.2 Ventilation Cooling of a HI-STORM System

4.0.3 Coupled Fluid Model of the HI-STORM System

4.2.1 Deleted

4.2.2 Deleted

4.2.3 Comparison of the Thermal Conductivity of METAMIC® and the CERMET Core of a BORAL® Neutron Absorber

4.3.1 Deleted

4.3.2 Deleted

4.3.3 Deleted

4.3.4 Deleted

4.4.1 Homogenization of the Storage Cell Cross-Section

4.4.2 MPC Cross-Section Replaced With an Equivalent Two Zone Axisymmetric Body

4.4.3 Westinghouse 17x17 OFA PWR Fuel Assembly Model

4.4.4 General Electric 9x9 BWR Fuel Assembly Model

4.4.5 Comparison of FLUENT Calculated Fuel Assembly Conductivity Results with Published Technical Data

4.4.6 Typical MPC Basket Parts in a Cross-Sectional View

4.4.7 "Box Wall-Neutron Absorber Sheathing" Sandwich

4.4.8 Deleted

4.4.9 MPC-24 Basket Cross-Section ANSYS Finite Element Model

4.4.10 MPC-68 Basket Cross-Section ANSYS Finite Element Model

4.4.11 Deleted

4.4.12 Deleted

4.4.13 Schematic Depiction of the HI-STORM Thermal Analysis

4.4.14 Deleted

4.4.15 Deleted

4.4.16 MPC-24 Peak Rod Axial Temperature Profile

4.4.17 MPC-68 Peak Rod Axial Temperature Profile

4.4.18 Deleted

4.4.19 MPC-24 Radial Temperature Profile (Hottest Basket Cross-Section)

4.4.20 MPC-68 Radial Temperature Profile (Hottest Basket Cross-Section)

4.4.21 Deleted

4.4.22 Deleted

4.4.23 Deleted

4.4.24 Illustration of Minimum Available Planar Area Per HI-STORM Module at an ISFSI

4.4.25 Fuel Basket Regionalized Loading Scenario

4.4.26 Bounding Overpack Annulus Axial Profiles

4.4.27 MPC-32 Regionalized Loading

4.4.28 MPC-24 & MPC-24E Regionalized Loading

4.4.29 MPC-68 Regionalized Loading

4.4.30 Bounding Basket Temperature Profile for Differential Expansion

4.4.31 Regionalized Storage Design Heat Load Curve for PWR MPCs

4.4.32 Regionalized Storage Design Heat Load Curve for BWR MPCs

4.4.33 Peak Cladding Temperature Variation in Regionalized Storage (MPC-32)

4.4.34 Peak Cladding Temperature Variation in Regionalized Storage (MPC-24/24E)

4.4.35 Peak Cladding Temperature Variation in Regionalized Storage (MPC-68)

4.5.1 Deleted

4.5.2 Deleted

4.5.3 Deleted

4.5.4 Peak Cladding Temperature in a Horizontal Configuration

4.A.1 Deleted

4.A.2 Deleted

4.A.3 Deleted

4.A.4 Deleted

4.A.5 Deleted

4.A.6 Deleted

4.A.7 Deleted

4.A.8 Deleted

4.A.9 Deleted

4.A.10 Deleted

4.A.11 Deleted

4.A.12 Deleted

4.A.13 Deleted

4.B.1 Cutaway View of a HI-STORM Overpack Standing on an ISFSI Pad

4.B.2 Depiction of the HI-STORM Ventilated Cask heat Dissipation Elements

4.B.3 Relative Significance of Heat Dissipation Elements in the HI-STORM 100

4.B.4 Air Access Restrictions in the HI-STORM Thermal Model

4.B.5 In-Plane Radiative Cooling of a HI-STORM Cask in an Array

4.B.6 In-Plane Radiative Blocking of a HI-STORM Cask by Hypothetical Reflecting Boundary

4.B.7 Radiative Heating of Reference HI-STORM Cask by Surrounding Casks

4.B.8 Deleted

5.1.1 Cross Section Elevation View of Overpack with Dose Point Location

5.1.2 Cross Section Elevation View of 125-Ton HI-TRAC Transfer Cask with Dose Point Locations

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)

5.1.4 Cross Section Elevation View of 100-Ton HI-TRAC Transfer Cask (With Pool Lid) With Dose Point Locations

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

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

5.1.7 Dose Rate 1-Foot From the Bottom of the Transfer Lid on the 100-Ton HI-TRAC Transfer Cask with the MPC-24 for 35,000 MWD/MTU and 5-Year Cooling

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

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)

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

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

5.1.12 Cross Section Elevation View of the HI-STORM 100S overpack with Dose Point Locations

5.3.1 HI-STORM 100 Overpack with MPC-32 Cross Sectional View as Modeled in MCNP

5.3.2 HI-STORM 100 Overpack with MPC-24 Cross Sectional View as Modeled in MCNP

5.3.3 HI-STORM 100 Overpack with MPC-68 Cross Sectional View as Modeled in MCNP

5.3.4 Cross Sectional View of an MPC-32 Basket Cell as Modeled in MCNP

5.3.5 Cross Sectional View of an MPC-24 Basket Cell as Modeled in MCNP

5.3.6 Cross Sectional View of an MPC-68 Basket Cell as Modeled in MCNP

5.3.7 HI-TRAC Overpack with MPC-24 Cross Sectional View as Modeled in MCNP

5.3.8 Axial Location of PWR Design Basis Fuel in the HI-STORM Overpack

5.3.9 Axial Location of BWR Design Basis Fuel in the HI-STORM Overpack

5.3.10 Cross Section of HI-STORM 100 Overpack

5.3.11 HI-STORM 100 Overpack Cross Sectional Elevation View

5.3.12 100-Ton HI-TRAC Transfer Cask with Pool Lid Cross Sectional Elevation View (As Modeled)

5.3.13 125-Ton HI-TRAC Transfer Cask with Pool Lid Cross Sectional Elevation View (As Modeled)

5.3.14 HI-TRAC 100 Transfer Cask Cross Sectional View (As Modeled)

5.3.15 HI-TRAC 125 Transfer Cask Cross Sectional View (As Modeled)

5.3.16 100-Ton HI-TRAC Transfer Lid (As Modeled)

5.3.17 125-Ton HI-TRAC Transfer Lid (As Modeled)

5.3.18 HI-STORM 100S Overpack Cross Sectional Elevation View

5.3.19 Gamma Shield Cross Plate Configuration of HI-STORM 100 and HI-STORM 100S

5.3.20 HI-TRAC 125D Transfer Cask Cross Sectional view (As Modeled)

5.3.21 Cross Sectional Views of the Current MPC-24 Design and the Superseded MPC-24 Which is Used in the MCNP Models

6.3.1 Typical Cell in the Calculation Model (Planar Cross-Section) with Representative Fuel in the MPC-24 Basket

6.3.1A Typical Cell in the Calculation Model (Planar Cross-Section) wth Representative Fuel in the MPC-24E Basket

6.3.2 Typical Cell in the Calculation Model (Planar Cross-Section) with Representative Fuel in the MPC-32 Basket

6.3.3 Typical Cell in the Calculation Model (Planar Cross-Section) with Representative Fuel in the MPC-68 Basket

6.3.4 Calculation Model (Planar Cross-Section) with Fuel Illustrated in One Quadrant of the MPC-24 and the MPC-24E

6.3.5 Calculation Model (Planar Cross-Section) with Fuel Illustrated in One Quadrant of the MPC-32

6.3.6 Calculation Model (Planar Cross-Section) with Fuel Illustrated in One Quadrant of the MPC-68

6.3.7 Sketch of the Calculational Model in the Axial Direction

6.4.1 Deleted

6.4.2 Failed Fuel Calculation Model (Planar Cross-Section) with 6x6 Array with 4 Missing Rods in the MPC-68 Basket

6.4.3 Failed Fuel Calculation Model (Planar Cross-Section) with 6x6 Array with 8 Missing Rods in the MPC-68 Basket

6.4.4 Failed Fuel Calculation Model (Planar Cross-Section) with 6x6 Array with 12 Missing Rods in the MPC-68 Basket

6.4.5 Failed Fuel Calculation Model (Planar Cross-Section) with 6x6 Array with 18 Missing Rods in the MPC-68 Basket

6.4.6 Failed Fuel Calculation Model (Planar Cross-Section) with 7x7 Array with 8 Missing Rods in the MPC-68 Basket

6.4.7 Failed Fuel Calculation Model (Planar Cross-Section) with 7x7 Array with 13 Missing Rods in the MPC-68 Basket

6.4.8 Failed Fuel Calculation Model (Planar Cross-Section) with 7x7 Array with 24 Missing Rods in the MPC-68 Basket

6.4.9 Failed Fuel Calculation Model (Planar Cross-Section) with Damaged Fuel Collapsed Into 8x8 Array in the MPC-68 Basket

6.4.10 Calculated K-Effective As A Function of Internal Moderator Density

6.4.11 Locations of the Damaged Fuel Container in the MPC-68 and MPC-68FF

6.4.12 Locations of the Damaged Fuel Containers in the MPC-24E

6.4.13 Maximum keff 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

6.4.14 Maximum keff for the MPC-24E with Generic PWR Damaged Fuel Container, Initial Enrichment of 4.0 wt% for Damaged and Intact Fuel

6.4.15 Thoria Rod Canister (Planar Cross-Section) with 18 Thoria Rods in the MPC-68 Basket

6.4.16 Locations of the Damaged Fuel Containers in the MPC-32

6.4.17 Damaged Fuel/Fuel Debris Calculation Model (Planar Cross-Section) with Bare Fuel Rods in the MPC-32 Basket)

6.A.1 MCNP4a Calculated k-eff Values for Various Values of the Spectral Index

6.A.2 KENO5a Calculated k-eff Values for Various Values of the Spectral Index

6.A.3 MCNP4a Calculated k-eff Values at Various U-235 Enrichments

6.A.4 KENO5a Calculated k-eff Values at Various U-235 Enrichments

6.A.5 Comparison of MCNP4a and KENO5a Calculations for Various Fuel Enrichments

6.A.6 Comparison of MCNP4a and KENO5a Calculations for Various Boron-10 Areal Densities

7.1.1 HI-STORM 100 System Confinement Boundary

8.1.1 Loading Operations Flow Diagram

8.1.2a Major HI-STORM 100 Loading Operations

8.1.2b Major HI-STORM 100 Loading Operations

8.1.2c Major HI-STORM 100 Loading Operations

8.1.2d Major HI-STORM 100 Loading Operations

8.1.2e Example of HI-STORM 100 Handling Options

8.1.2.f Example of HI-TRAC Handling Options (Missile Shields Not Shown for Clarity)

8.1.3 Lift Yoke Engagement and Vertical HI-TRAC Handling (Shown with the Pool Lid and the Transfer Lid)

8.1.4 HI-TRAC Upending/Downending in the Transfer Frame

8.1.5 HI-STORM Vertical Handling

8.1.6 MPC Upending in the MPC Upending Frame

8.1.7 MPC Rigging for Vertical Lifts

8.1.8 MPC Alignment in HI-TRAC

8.1.9 MPC Lid and HI-TRAC Accessory Rigging

8.1.10 Fuel Spacers

8.1.11 Drain Port Details

8.1.12 Drain Line Positioning

8.1.13 Annulus Shield/Annulus Seal

8.1.14 Annulus Overpressure System

8.1.15 HI-TRAC Lid Retention System in Exploded View

8.1.16 MPC Vent and Drain Port RVOA Connector

8.1.17 Drain Line Installation

8.1.18 Temporary Shield Ring

8.1.19 MPC Water Pump-Down for MPC Lid Welding Operations

8.1.20 MPC Air Displacement and Hydrostatic Testing

8.1.21 MPC Blowdown

8.1.22a Vacuum Drying System

8.1.22b Moisture Removal System

8.1.23 Helium Backfill System

8.1.24 MPC Lift Cleats

8.1.25 MPC Support Stays

8.1.26 HI-TRAC Bottom Lid Replacement

8.1.27 HI-STORM Lid Rigging

8.1.28 Sample MPC Transfer Options

8.1.29a Sample HI-STORM and HI-TRAC Transfer Options

8.1.29b Sample HI-STORM and HI-TRAC Transfer Options

8.1.30 Sample HI-STORM Vent Duct Shield Inserts

8.1.31 HI-TRAC Alignment Over HI-STORM

8.1.32 Examples of an MPC Downloader

8.1.33 Transfer Lid Trim Plates

8.1.34a HI-STORM Vent Screens and Gamma Shield Cross Plate Installation (Typ.)

8.1.34b HI-STORM Thermocouple Installation

8.1.35 HI-STORM Placement of the ISFSI Pad

8.1.36 HI-STORM Jacking

8.1.37 HI-TRAC Lid Bolt Torquing Pattern

8.3.1 Unloading Operations Flow Diagram

8.3.2a Major HI-STORM 100 Unloading Operations

8.3.2b Major HI-STORM 100 Unloading Operations

8.3.2c Major HI-STORM 100 Unloading Operations

8.3.2d Major HI-STORM 100 Unloading Operations

8.3.3 MPC Gas Sampling in Preparation for Unloading

8.3.4	MPC Cool-Down
8.4.1a	HI-STAR and HI-TRAC Mating
8.4.1b	HI-STAR and HI-TRAC 125D Mating
8.5.1	HI-STAR Annulus Gas Sampling
10.1.1	HI-STORM 100 System Auxiliary/Temporary Shielding
10.3.1a	Operator Work Locations Used for Estimating Personnel Exposure
10.3.1b	Operator Work Locations Used for Estimating Personnel Exposure
10.3.1c	Operator Work Locations Used for Estimating Personnel Exposure
10.3.1d	Operator Work Locations Used for Estimating Personnel Exposure
10.3.1e	Operator Work Locations Used for Estimating Personnel Exposure
11.2.1	Fire Transient ANSYS Model Element Plot
11.2.2	Deleted
11.2.3	Deleted
11.2.4	Deleted
11.2.5	Deleted
11.2.6	Allowable Burial Under Debris Time Versus Decay Heat Load
11.2.7	Deleted
11.2.8	Peak Cladding Temperature in a Horizontal HI-TRAC

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
i	2C			
ii	2C			
iii	2C			
iv	2C			
v	2C			
vi	2C			
vii	2C			
viii	2C			
ix	2C			
x	2C			
xi	2C			
xii	2C			
xiii	2C			
xiv	2C			
xv	2C			
xvi	2C			
xvii	2C			
xviii	2C			
xix	2C			
xx	2C			
xxi	2C			
xxii	2C			
xxiii	2C			
xxiv	2C			
xxv	2C			
xxvi	2C			
xxvii	2C			
xxviii	2C			
xxix	2C			
xxx	2C			
xxxi	2C			
xxxii	2C			
xxxiii	2C			
xxxiv	2C			
xxxv	2C			
xxxvi	2C			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
1.0-1	2B		1.2-10	2B
1.0-2	2B		1.2-11	2B
1.0-3	2B		1.2-12	2B
1.0-4	2B		1.2-13	2C
1.0-5	2B		1.2-14	2C
1.0-6	2B		1.2-15	2C
1.0-7	2B		1.2-16	2C
1.0-8	2B		1.2-17	2C
1.0-9	2B		1.2-18	2C
1.0-10	2B		1.2-19	2B
1.0-11	2B		1.2-20	2B
1.0-12	2B		1.2-21	2B
1.0-13	2C		1.2-22	2B
1.0-14	2C		1.2-23	2B
1.0-15	2B		1.2-24	2B
1.0-16	2B		1.2-25	2C
1.0-17	2B		1.2-26	2C
1.0-18	2B		1.2-27	2C
1.0-19	2B		1.2-28	2C
1.0-20	2B		1.2-29	2C
1.0-21	2B		1.2-30	2C
1.0-22	2B		1.2-31	2B
1.0-23	2B		1.2-32	2B
1.0-24	2B		1.2-33	2B
1.0-25	2C		1.2-34	2B
1.0-26	2C		1.2-35	2B
1.0-27	2C		1.2-36	2B
1.0-28	2C		1.2-37	2B
1.0-29	2C		1.2-38	2B
1.0-30	2C		1.2-39	2B
1.0-31	2B		1.2-40	2B
1.1-1	2C		1.2-41	2B
1.1-2	2C		Fig. 1.2.1	1
1.1-3	2A		Fig. 1.2.1A	1
1.1-4	2A		Fig. 1.2.2	1
Fig. 1.1.1	0		Fig. 1.2.3	2B
Fig. 1.1.1A	1		Fig. 1.2.4	1
Fig. 1.1.2	0		Fig. 1.2.5	0
Fig. 1.1.3	1		Fig. 1.2.6	0
Fig. 1.1.3A	1		Fig. 1.2.7	1
Fig. 1.1.4	1		Fig. 1.2.8	1
Fig. 1.1.5	1		Fig. 1.2.8A	1
1.2-1	2B		Fig. 1.2.9	1
1.2-2	2B		Fig. 1.2.10	0
1.2-3	2C		Fig. 1.2.11	0
1.2-4	2C		Fig. 1.2.12	0
1.2-5	2B		Fig. 1.2.16a	0
1.2-6	2B		Fig. 1.2.16b	0
1.2-7	2B		Fig. 1.2.16c	0
1.2-8	2B		Fig. 1.2.16d	0
1.2-9	2B		Fig. 1.2.16e	0

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
Fig. 1.2.16f	0			
Fig. 1.2.17a	0			
Fig. 1.2.17b	0			
Fig. 1.2.17c	0			
Fig. 1.2.17d	0			
Fig. 1.2.18	1			
1.3-1	2B			
1.4-1	1			
1.4-2	1			
1.4-3	1			
Fig. 1.4.1	0			
Fig. 1.4.2	0			
1.5-1	1			
1.5-2	1			
Drawings	See Section 1.5			
Bills-of-Material	See Section 1.5			
1.6-1	2C			
1.6-2	2C			
1.A-1	1			
1.A-2	1			
1.A-3	1			
1.A-4	1			
1.A-5	1			
1.A-6	1			
1.A-7	1			
Fig. 1.A.1	0			
Fig. 1.A.2	0			
Fig. 1.A-3	0			
Fig. 1.A.4	0			
Fig. 1.A.5	0			
1.B-1	2B			
1.B-2	2B			
1.B-3	2B			
1.C-1	1			
1.C-3	0			
1.C-4	0			
1.D-1	2B			
1.D-2	2B			
1.D-3	2C			
1.D-4	2C			
1.D-5	2C			
1.D-6	2C			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
2.0-1	2C		2.1-14	2B
2.0-2	2C		2.1-15	2B
2.0-3	2C		2.1-16	2B
2.0-4	2C		2.1-17	2B
2.0-5	2C		2.1-18	2B
2.0-6	2C		2.1-19	2B
2.0-7	2C		2.1-20	2B
2.0-8	2C		2.1-21	2B
2.0-9	2C		2.1-22	2B
2.0-10	2C		2.1-23	2B
2.0-11	2B		2.1-24	2B
2.0-12	2B		2.1-25	2B
2.0-13	2B		2.1-26	2B
2.0-14	2B		2.1-27	2B
2.0-15	2B		2.1-28	2B
2.0-16	2B		2.1-29	2B
2.0-17	2B		2.1-30	2B
2.0-18	2B		2.1-31	2B
2.0-19	2C		2.1-32	2B
2.0-20	2C		2.1-33	2B
2.0-21	2B		2.1-34	2C
2.0-22	2B		2.1-35	2C
2.0-23	2B		2.1-36	2C
2.0-24	2B		2.1-37	2C
2.0-25	2B		2.1-38	2C
2.0-26	2B		2.1-39	2C
2.0-27	2C		2.1-40	2C
2.0-28	2C		2.1-41	2C
2.0-29	2C		2.1-42	2C
2.0-30	2C		2.1-43	2C
2.0-31	2B		2.1-44	2C
2.0-32	2B		2.1-45	2C
2.0-33	2B		2.1-46	2C
2.0-34	2B		2.1-47	2C
2.0-35	2B		2.1-48	2C
2.0-36	2B		2.1-49	2C
2.0-37	2B		2.1-50	2C
2.0-38	2B		2.1-51	2C
2.1-1	2B		2.1-52	2C
2.1-2	2B		2.1-53	2B
2.1-3	2B		2.1-54	2B
2.1-4	2B		2.1-55	2B
2.1-5	2B		2.1-56	2B
2.1-6	2B		2.1-57	2B
2.1-7	2C		2.1-58	2B
2.1-8	2C		2.1-59	2B
2.1-9	2C		2.1-60	2B
2.1-10	2C		2.1-61	2B
2.1-11	2B		2.1-62	2B
2.1-12	2B		2.1-63	2B
2.1-13	2B		2.1-64	2B

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
2.1-65	2B		2.2-36	2B
2.1-66	2B		2.2-37	2B
2.1-67	2B		2.2-38	2B
2.1-68	2B		2.2-39	2B
2.1-69	2B		2.2-40	2B
2.1-70	2B		2.2-41	2B
Fig. 2.1.1	2B		2.2-42	2B
Fig. 2.1.2	1		2.2-43	2C
Fig. 2.1.2A	1		2.2-44	2C
Fig. 2.1.2B	1		2.2-45	2C
Fig. 2.1.2C	1		2.2-46	2C
Fig. 2.1.2D	2B		2.2-47	2B
Fig. 2.1.3	0		2.2-48	2B
Fig. 2.1.4	0		2.2-49	2B
Fig. 2.1.5	0		2.2-50	2B
Fig. 2.1.9	2A		2.2-51	2B
2.2-1	2C		2.2-52	2B
2.2-2	2C		2.2-53	2B
2.2-3	2C		2.2-54	2B
2.2-4	2C		2.2-55	2B
2.2-5	2C		2.2-56	2B
2.2-6	2C		2.2-57	2B
2.2-7	2C		2.3-1	2C
2.2-8	2C		2.3-2	2C
2.2-9	2C		2.3-3	2C
2.2-10	2C		2.3-4	2C
2.2-11	2B		2.3-5	2C
2.2-12	2B		2.3-6	2C
2.2-13	2B		2.3-7	2B
2.2-14	2B		2.3-8	2B
2.2-15	2B		2.3-9	2B
2.2-16	2B		2.3-10	2B
2.2-17	2B		2.3-11	2B
2.2-18	2B		2.3-12	2B
2.2-19	2B		2.3-13	2B
2.2-20	2B		2.3-14	2B
2.2-21	2C		2.3-15	2B
2.2-22	2C		2.3-16	2B
2.2-23	2B		2.3-17	2B
2.2-24	2B		2.3-18	2B
2.2-25	2B		2.3-19	2B
2.2-26	2B		2.3-20	2B
2.2-27	2B		2.3-21	2B
2.2-28	2B		2.3-22	2B
2.2-29	2B		2.3-23	2B
2.2-30	2B		Fig. 2.3.1	0
2.2-31	2B		Fig. 2.3.2	0
2.2-32	2B		Fig. 2.3.3	0
2.2-33	2B		Fig. 2.3.4	0
2.2-34	2B			
2.2-35	2B			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
2.4-1	2A			
2.4-2	2A			
2.4-3	2A			
2.5-1	0			
2.6-1	2C			
2.6-2	2C			
2.6-3	2C			
2A-1	1			
2A-2	1			
2A-3	1			
2A-4	1			
2A-5	1			
Fig. 2.A.1	1			
2.B-1	2B			
2.B-2	2B			
2.B-3	2B			
2.B-4	2B			
Fig. 2.B.1	1			
2.C-1	2C			
2.C-2	2C			
Fig. 2.C.1	2C			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision	Page	Revision
3.0-1	2B	3.1-43	2B
3.0-2	2B	Fig. 3.1.1	1
3.0-3	2B	Fig. 3.1.2	1
3.0-4	2B	Fig. 3.1.3	1
3.0-5	2B	3.2-1	1
3.0-6	2B	3.2-2	1
3.0-7	2B	3.2-3	1
3.0-8	2B	3.2-4	1
3.0-9	2B	3.2-5	1
3.0-10	2B	3.2-6	1
3.1-1	2B	3.2-7	1
3.1-2	2B	3.2-8	1
3.1-3	2B	3.2-9	1
3.1-4	2B	3.2-10	1
3.1-5	2B	3.2-11	1
3.1-6	2B	3.2-12	1
3.1-7	2B	3.3-1	1
3.1-8	2B	3.3-2	1
3.1-9	2B	3.3-3	1
3.1-10	2B	3.3-4	1
3.1-11	2B	3.3-5	1
3.1-12	2B	3.3-6	1
3.1-13	2B	3.3-7	1
3.1-14	2B	3.3-8	1
3.1-15	2B	3.3-9	1
3.1-16	2B	3.3-10	Deleted
3.1-17	2B	3.4-1	2B
3.1-18	2B	3.4-2	2B
3.1-19	2B	3.4-3	2B
3.1-20	2B	3.4-4	2B
3.1-21	2B	3.4-5	2B
3.1-22	2B	3.4-6	2B
3.1-23	2B	3.4-7	2B
3.1-24	2B	3.4-8	2B
3.1-25	2B	3.4-9	2B
3.1-26	2B	3.4-10	2B
3.1-27	2B	3.4-11	2B
3.1-28	2B	3.4-12	2B
3.1-29	2B	3.4-13	2B
3.1-30	2B	3.4-14	2B
3.1-31	2B	3.4-15	2B
3.1-32	2B	3.4-16	2B
3.1-33	2B	3.4-17	2B
3.1-34	2B	3.4-18	2B
3.1-35	2B	3.4-19	2B
3.1-36	2B	3.4-20	2B
3.1-37	2B	3.4-21	2B
3.1-38	2B	3.4-22	2B
3.1-39	2B	3.4-23	2B
3.1-40	2B	3.4-24	2B
3.1-41	2B	3.4-25	2B
3.1-42	2B	3.4-26	2B

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
3.4-27	2B		3.4-79	2B
3.4-28	2B		3.4-80	2B
3.4-29	2B		3.4-81	2B
3.4-30	2B		3.4-82	2B
3.4-31	2B		3.4-83	2B
3.4-32	2B		3.4-84	2B
3.4-33	2B		3.4-85	2B
3.4-34	2B		3.4-86	2B
3.4-35	2B		3.4-87	2B
3.4-36	2B		3.4-88	2B
3.4-37	2B		3.4-89	2B
3.4-38	2B		3.4-90	2B
3.4-39	2B		3.4-91	2B
3.4-40	2B		3.4-92	2B
3.4-41	2B		3.4-93	2B
3.4-42	2B		3.4-94	2B
3.4-43	2B		3.4-95	2B
3.4-44	2B		3.4-96	2B
3.4-45	2B		3.4-97	2B
3.4-46	2B		3.4-98	2B
3.4-47	2B		3.4-99	2B
3.4-48	2B		3.4-100	2B
3.4-49	2B		3.4-101	2B
3.4-50	2B		3.4-102	2B
3.4-51	2B		3.4-103	2B
3.4-52	2B		3.4-104	2B
3.4-53	2B		3.4-105	2B
3.4-54	2B		3.4-106	2B
3.4-55	2B		3.4-107	2B
3.4-56	2B		3.4-108	2B
3.4-57	2B		3.4-109	2B
3.4-58	2B		3.4-110	2B
3.4-59	2B		3.4-111	2B
3.4-60	2B		3.4-112	2B
3.4-61	2B		3.4-113	2B
3.4-62	2B		3.4-114	2B
3.4-63	2B		3.4-115	2B
3.4-64	2B		3.4-116	2B
3.4-65	2B		3.4-117	2B
3.4-66	2B		3.4-118	2B
3.4-67	2B		3.4-119	2B
3.4-68	2B		3.4-120	2B
3.4-69	2B		3.4-121	2B
3.4-70	2B		3.4-122	2B
3.4-71	2B		3.4-123	2B
3.4-72	2B		3.4-124	2B
3.4-73	2B		3.4-125	2B
3.4-74	2B		3.4-126	2B
3.4-75	2B		Fig. 3.4.1	0
3.4-76	2B		Fig. 3.4.2	1
3.4-77	2B		Fig. 3.4.3	0
3.4-78	2B			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision	Page	Revision
Fig. 3.4.4	0	3.5-4	0
Fig. 3.4.5	1	3.5-5	0
Fig. 3.4.6	0	3.5-6	0
Fig. 3.4.7	0	3.5-7	0
Fig. 3.4.8	0	3.5-8	0
Fig. 3.4.9	0	3.5-9	0
Fig. 3.4.10	1	3.5-10	0
Fig. 3.4.11	0	3.5-11	0
Fig. 3.4.12	0	3.5-12	0
Fig. 3.4.13	0	3.5-13	0
Fig. 3.4.14	0	3.5-14	0
Fig. 3.4.15	0	3.5-15	0
Fig. 3.4.16	0	3.5-16	0
Fig. 3.4.16a	0	3.5-17	0
Fig. 3.4.16b	0	3.5-18	0
Fig. 3.4.17	0	3.5-19	0
Fig. 3.4.18	0	Fig. 3.5.1	0
Fig. 3.4.19	0	Fig. 3.5.2	0
Fig. 3.4.20	0	Fig. 3.5.3	0
Fig. 3.4.21	0	Fig. 3.5.4	0
Fig. 3.4.22	0	Fig. 3.5.5	0
Fig. 3.4.23	0	Fig. 3.5.6	0
Fig. 3.4.24	0	Fig. 3.5.7	0
Fig. 3.4.25	0	Fig. 3.5.8	0
Fig. 3.4.26	0	Fig. 3.5.9	0
Fig. 3.4.27	0	3.6-1	2A
Fig. 3.4.28	0	3.6-2	2A
Fig. 3.4.29	0	3.6-3	2A
Fig. 3.4.30	1	3.6-4	2A
Fig. 3.4.31	1	3.6-5	2A
Fig. 3.4.32	1	3.6-6	2A
Fig. 3.4.33	1	3.6-7	2A
Fig. 3.4.34	1	3.6-8	2A
Fig. 3.4.35	1	3.6-9	2A
Fig. 3.4.36	1	3.7-1	2A
Fig. 3.4.37	1	3.7-2	2A
Fig. 3.4.38	1	3.7-3	2A
Fig. 3.4.39	1	3.7-4	2A
Fig. 3.4.40	1	3.7-5	2A
Fig. 3.4.41	1	3.7-6	2A
Fig. 3.4.42	1	3.7-7	2A
Fig. 3.4.43	1	3.7-8	2A
Fig. 3.4.44	1	3.7-9	2A
Fig. 3.4.45	1	3.7-10	2A
Fig. 3.4.46	1	3.8-1	1
Fig. 3.4.47	1	3.8-2	1
Fig. 3.4-48	1	3.A-1	1
Fig. 3.4-49	1	3.A-2	1
3.5-1	0	3.A-3	1
3.5-2	0	3.A-4	1
3.5-3	0	3.A-5	1

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
3.A-6	1		3.B-24	Deleted
3.A-7	1		3.B-25	Deleted
3.A-8	1		3.B-26	Deleted
3.A-9	1		3.B-27	Deleted
3.A-10	1		3.B-28	Deleted
3.A-11	1		3.B-29	Deleted
3.A-12	1		3.B-30	Deleted
3.A-13	1		3.B-31	Deleted
3.A-14	1		3.B-32	Deleted
3.A-15	1		3.B-33	Deleted
Fig. 3.A.1	0		3.B-34	Deleted
Fig. 3.A.2	0		3.B-35	Deleted
Fig. 3.A.3	0		3.B-36	Deleted
Fig. 3.A.4	0		3.B-37	Deleted
Fig. 3.A.5	0		3.B-38	Deleted
Fig. 3.A.6	0		3.B-39	Deleted
Fig. 3.A.7	0		3.B-40	Deleted
Fig. 3.A.8	0		3.B-41	Deleted
Fig. 3.A.9	0		3.B-42	Deleted
Fig. 3.A.10	0		3.B-43	Deleted
Fig. 3.A.11	0		3.B-44	Deleted
Fig. 3.A.12	0		3.B-45	Deleted
Fig. 3.A.13	0		3.B-46	Deleted
Fig. 3.A.14	0		3.B-47	Deleted
Fig. 3.A.15	0		3.B-48	Deleted
Fig. 3.A.16	0		3.B-49	Deleted
Fig. 3.A.17	0		3.B-50	Deleted
Fig. 3.A.18	0		3.B-51	Deleted
3.B-1	Deleted		3.B-52	Deleted
3.B-2	Deleted		3.B-53	Deleted
3.B-3	Deleted		3.B-54	Deleted
3.B-4	Deleted		3.B-55	Deleted
3.B-5	Deleted		3.B-56	Deleted
3.B-6	Deleted		3.B-57	Deleted
3.B-7	Deleted		3.B-58	Deleted
3.B-8	Deleted		3.B-59	Deleted
3.B-9	Deleted		3.B-60	Deleted
3.B-10	Deleted		3.B-61	Deleted
3.B-11	Deleted		3.B-62	Deleted
3.B-12	Deleted		3.C-1	Deleted
3.B-13	Deleted		3.C-2	Deleted
3.B-14	Deleted		3.C-3	Deleted
3.B-15	Deleted		3.C-4	Deleted
3.B-16	Deleted		3.C-5	Deleted
3.B-17	Deleted		3.C-6	Deleted
3.B-18	Deleted		3.C-7	Deleted
3.B-19	Deleted		3.C-8	Deleted
3.B-20	Deleted		Fig. 3.C.1	Deleted
3.B-21	Deleted		Fig. 3.C.2	Deleted
3.B-22	Deleted		Fig. 3.C.3	Deleted
3.B-23	Deleted		3.D-1	Deleted
			3.D-2	Deleted

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision	Page	Revision
3.D-3	Deleted	3.G-8	Deleted
3.D-4	Deleted	3.G-9	Deleted
3.D-5	Deleted	3.G-10	Deleted
3.D-6	Deleted	3.G-11	Deleted
3.D-7	Deleted	3.G-12	Deleted
3.D-8	Deleted	3.G-13	Deleted
3.D-9	Deleted	Fig. 3.G.1	Deleted
3.D-10	Deleted	Fig. 3.G.2	Deleted
3.D-11	Deleted	Fig. 3.G.3	Deleted
3.D-12	Deleted	Fig. 3.G.4	Deleted
3.D-13	Deleted	Fig. 3.G.5	Deleted
3.D-14	Deleted	3.H-1	Deleted
3.D-15	Deleted	3.H-2	Deleted
Fig. 3.D.1	Deleted	3.H-3	Deleted
Fig. 3.D.2a	Deleted	3.H-4	Deleted
Fig. 3.D.2b	Deleted	3.H-5	Deleted
Fig. 3.D.2c	Deleted	3.H-6	Deleted
Fig. 3.D.3	Deleted	3.H-7	Deleted
Fig. 3.D.4a	Deleted	Fig. 3.H.1	Deleted
Fig. 3.D.4b	Deleted	3.I-1	Deleted
Fig. 3.D.4c	Deleted	3.I-2	Deleted
Fig. 3.D.5a	Deleted	3.I-3	Deleted
Fig. 3.D.5b	Deleted	3.I-4	Deleted
Fig. 3.D.5c	Deleted	3.I-5	Deleted
3.E-1	Deleted	3.I-6	Deleted
3.E-2	Deleted	3.I-7	Deleted
3.E-3	Deleted	3.I-8	Deleted
3.E-4	Deleted	3.I-9	Deleted
3.E-5	Deleted	3.I-10	Deleted
3.E-6	Deleted	Fig. 3.I.1	Deleted
3.E-7	Deleted	Appendix 3.J	Deleted
3.E-8	Deleted	3.K-1	Deleted
3.E-9	Deleted	3.K-2	Deleted
3.E-10	Deleted	3.K-3	Deleted
Fig. 3.E.1	Deleted	3.K-4	Deleted
Fig. 3.E.2	Deleted	3.K-5	Deleted
Fig. 3.E.3	Deleted	3.K-6	Deleted
3.F-1	Deleted	3.K-7	Deleted
3.F-2	Deleted	3.L-1	Deleted
3.F-3	Deleted	3.L-2	Deleted
3.F-4	Deleted	3.L-3	Deleted
Fig. 3.F.1	Deleted	3.L-4	Deleted
Fig. 3.F.2	Deleted	3.L-5	Deleted
Fig. 3.F.3	Deleted	3.L-6	Deleted
Fig. 3.F.4	Deleted	3.L-7	Deleted
3.G-1	Deleted	3.L-8	Deleted
3.G-2	Deleted	3.L-9	Deleted
3.G-3	Deleted	3.L-10	Deleted
3.G-4	Deleted	3.L-11	Deleted
3.G-5	Deleted	3.L-12	Deleted
3.G-6	Deleted	3.L-13	Deleted
3.G-7	Deleted	3.M-1	Deleted

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
3.M-2	Deleted		3.W-6	Deleted
3.M-3	Deleted		3.W-7	Deleted
3.M-4	Deleted		3.W-8	Deleted
3.M-5	Deleted		3.W-9	Deleted
3.M-6	Deleted		3.W-10	Deleted
3.M-7	Deleted		Fig. 3.W.1	Deleted
3.M-8	Deleted		3.X-1	Deleted
3.M-9	Deleted		3.X-2	Deleted
3.M-10	Deleted		3.X-3	Deleted
3.M-11	Deleted		3.X-4	Deleted
3.M-12	Deleted		3.X-5	Deleted
3.M-13	Deleted		3.X-6	Deleted
3.M-14	Deleted		3.X-7	Deleted
3.M-15	Deleted		3.X-8	Deleted
3.M-16	Deleted		3.X-9	Deleted
3.M-17	Deleted		3.X-10	Deleted
3.M-18	Deleted		Fig. 3.X.1	Deleted
3.M-19	Deleted		Fig. 3.X.2	Deleted
Appendix 3.N	Deleted in Rev 1		Fig. 3.X.3	Deleted
Appendix 3.O	Deleted in Rev 1		Fig. 3.X.4	Deleted
Appendix 3.P	Deleted in Rev 1		Fig. 3.X.5	Deleted
Appendix 3.Q	Deleted in Rev 1		3.Y-1	Deleted
Appendix 3.R	Deleted in Rev 1		3.Y-2	Deleted
Appendix 3.S	Deleted in Rev 1		3.Y-3	Deleted
Appendix 3.T	Deleted in Rev 1		3.Y-4	Deleted
3.U-1	Deleted		3.Y-5	Deleted
3.U-2	Deleted		3.Y-6	Deleted
3.U-3	Deleted		3.Y-7	Deleted
3.U-4	Deleted		3.Y-8	Deleted
3.U-5	Deleted		3.Y-9	Deleted
3.U-6	Deleted		3.Y-10	Deleted
3.U-7	Deleted		3.Y-11	Deleted
3.U-8	Deleted		3.Y-12	Deleted
3.U-9	Deleted		3.Y-13	Deleted
3.U-10	Deleted		3.Y-14	Deleted
Fig. 3.U.1	Deleted		3.Y-15	Deleted
3.V-1	Deleted		3.Y-16	Deleted
3.V-2	Deleted		3.Y-17	Deleted
3.V-3	Deleted		3.Y-18	Deleted
3.V-4	Deleted		3.Y-19	Deleted
3.V-5	Deleted		3.Y-20	Deleted
3.V-6	Deleted		Fig. 3.Y.1	Deleted
3.V-7	Deleted		Fig. 3.Y.2	Deleted
3.V-8	Deleted		3.Z-1	Deleted
3.V-9	Deleted		3.Z-2	Deleted
3.V-10	Deleted		3.Z-3	Deleted
Fig. 3.V.1	Deleted		3.Z-4	Deleted
3.W-1	Deleted		3.Z-5	Deleted
3.W-2	Deleted		3.Z-6	Deleted
3.W-3	Deleted		3.Z-7	Deleted
3.W-4	Deleted		3.Z-8	Deleted
3.W-5	Deleted		3.Z-9	Deleted

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision	Page	Revision
3.Z-10	Deleted	3.AD-2	Deleted
3.Z-11	Deleted	3.AD-3	Deleted
3.Z-12	Deleted	3.AD-4	Deleted
Fig. 3.Z.1	Deleted	3.AD-5	Deleted
Fig. 3.Z.2	Deleted	3.AD-6	Deleted
Fig. 3.Z.3	Deleted	3.AD-7	Deleted
Fig. 3.Z.4	Deleted	3.AD-8	Deleted
Fig. 3.Z.5	Deleted	3.AD-9	Deleted
Fig. 3.Z.6	Deleted	3.AD-10	Deleted
3.AA-1	Deleted	3.AD-11	Deleted
3.AA-2	Deleted	3.AD-12	Deleted
3.AA-3	Deleted	3.AD-13	Deleted
3.AA-4	Deleted	3.AD-14	Deleted
3.AA-5	Deleted	3.AD-15	Deleted
3.AA-6	Deleted	3.AD-16	Deleted
3.AA-7	Deleted	3.AD-17	Deleted
3.AA-8	Deleted	3.AD-18	Deleted
Fig. 3.AA.1	Deleted	3.AD-19	Deleted
Fig. 3.AA.2	Deleted	Fig. 3.AD.1	Deleted
Fig. 3.AA.3	Deleted	Fig. 3.AD.2	Deleted
Fig. 3.AA.4	Deleted	Fig. 3.AD.3	Deleted
Fig. 3.AA.5	Deleted	3.AE-1	Deleted
Fig. 3.AA.6	Deleted	3.AE-2	Deleted
Fig. 3.AA.7	Deleted	3.AE-3	Deleted
Fig. 3.AA.8	Deleted	3.AE-4	Deleted
3.AB-1	Deleted	3.AE-5	Deleted
3.AB-2	Deleted	3.AE-6	Deleted
3.AB-3	Deleted	3.AE-7	Deleted
3.AB-4	Deleted	Fig. 3.AE.1	Deleted
3.AB-5	Deleted	Fig. 3.AE.1b	Deleted in Rev. 1
3.AB-6	Deleted	Fig. 3.AE.1c	Deleted in Rev. 1
3.AB-7	Deleted	Fig. 3.AE.2	Deleted
3.AB-8	Deleted	Fig. 3.AE.3	Deleted
3.AB-9	Deleted	Fig. 3.AE.4	Deleted in Rev. 1
3.AB-10	Deleted	3.AF-1	Deleted
3.AB-11	Deleted	3.AF-2	Deleted
3.AB-12	Deleted	3.AF-3	Deleted
3.AB-13	Deleted	3.AF-4	Deleted
3.AB-14	Deleted	3.AF-5	Deleted
3.AC-1	Deleted	3.AF-6	Deleted
3.AC-2	Deleted	3.AF-7	Deleted
3.AC-3	Deleted	3.AF-8	Deleted
3.AC-4	Deleted	3.AG-1	Deleted
3.AC-5	Deleted	3.AG-2	Deleted
3.AC-6	Deleted	3.AG-3	Deleted
3.AC-7	Deleted	3.AG-4	Deleted
3.AC-8	Deleted	3.AG-5	Deleted
3.AC-9	Deleted	3.AG-6	Deleted
3.AC-10	Deleted	3.AG-7	Deleted
3.AC-11	Deleted	3.AG-8	Deleted
3.AC-12	Deleted	3.AG-9	Deleted
3.AD-1	Deleted	3.AG-10	Deleted

Page	Revision		Page	Revision
3.AG-11	Deleted		3.AJ-5	Deleted
3.AG-12	Deleted		3.AJ-6	Deleted
3.AG-13	Deleted		3.AJ-7	Deleted
3.AG-14	Deleted		3.AJ-8	Deleted
3.AG-15	Deleted		3.AJ-9	Deleted
3.AG-16	Deleted		3.AJ-10	Deleted
3.AG-17	Deleted		3.AJ-11	Deleted
3.AG-18	Deleted		3.AJ-12	Deleted
3.AG-19	Deleted		3.AJ-13	Deleted
3.AG-20	Deleted		3.AJ-14	Deleted
3.AG-21	Deleted		3.AJ-15	Deleted
3.AG-22	Deleted		3.AJ-16	Deleted
3.AG-23	Deleted		3.AJ-17	Deleted
3.AG-24	Deleted		3.AJ-18	Deleted
3.AH-1	Deleted		3.AJ-19	Deleted
3.AH-2	Deleted		Fig. 3.AJ.1	Deleted
3.AH-3	Deleted		Fig. 3.AJ.2	Deleted
3.AH-4	Deleted		Fig. 3.AJ.3	Deleted
3.AH-5	Deleted		3.AK-1	Deleted
3.AH-6	Deleted		3.AK-2	Deleted
3.AH-7	Deleted		3.AK-3	Deleted
3.AH-8	Deleted		3.AK-4	Deleted
3.AH-9	Deleted		3.AK-5	Deleted
3.AI-1	Deleted		3.AK-6	Deleted
3.AI-2	Deleted		3.AK-7	Deleted
3.AI-3	Deleted		3.AK-8	Deleted
3.AI-4	Deleted		3.AK-9	Deleted
3.AI-5	Deleted		3.AK-10	Deleted
3.AI-6	Deleted		3.AK-11	Deleted
3.AI-7	Deleted		3.AK-12	Deleted
3.AI-8	Deleted		3.AK-13	Deleted
3.AI-9	Deleted		3.AK-14	Deleted
3.AI-10	Deleted		3.AK-15	Deleted
3.AI-11	Deleted		3.AK-16	Deleted
3.AI-12	Deleted		3.AK-17	Deleted
3.AI-13	Deleted		3.AK-18	Deleted
3.AI-14	Deleted		3.AL-1	Deleted
3.AI-15	Deleted		3.AL-2	Deleted
3.AI-16	Deleted		3.AL-3	Deleted
3.AI-17	Deleted		3.AL-4	Deleted
3.AI-18	Deleted		3.AL-5	Deleted
3.AI-19	Deleted		3.AL-6	Deleted
Fig. 3.AI.1	Deleted		3.AL-7	Deleted
Fig. 3.AI.2	Deleted		3.AL-8	Deleted
Fig. 3.AI.3	Deleted		3.AL-9	Deleted
Fig. 3.AI.4	Deleted		3.AL-10	Deleted
Fig. 3.AI.5	Deleted		3.AM-1	Deleted
Fig. 3.AI.6	Deleted		3.AM-2	Deleted
3.AJ-1	Deleted		3.AM-3	Deleted
3.AJ-2	Deleted		3.AM-4	Deleted
3.AJ-3	Deleted		3.AM-5	Deleted
3.AJ-4	Deleted		3.AM-6	Deleted

Page	Revision		Page	Revision
3.AM-7	Deleted		Fig. 3.AN.15	Deleted
3.AM-8	Deleted		Fig. 3.AN.16	Deleted
3.AM-9	Deleted		Fig. 3.AN.17	Deleted
3.AM-10	Deleted		Fig. 3.AN.18	Deleted
3.AM-11	Deleted		Fig. 3.AN.19	Deleted
3.AM-12	Deleted		Fig. 3.AN.20	Deleted
3.AM-13	Deleted		Fig. 3.AN.21	Deleted
3.AM-14	Deleted		Fig. 3.AN.22	Deleted
3.AM-15	Deleted		Fig. 3.AN.23	Deleted
3.AM-16	Deleted		Fig. 3.AN.24	Deleted
3.AM-17	Deleted		Fig. 3.AN.25	Deleted
3.AM-18	Deleted		Fig. 3.AN.26	Deleted
3.AM-19	Deleted		Fig. 3.AN.27	Deleted
3.AM-20	Deleted		Fig. 3.AN.28	Deleted
3.AM-21	Deleted		Fig. 3.AN.29	Deleted
3.AM-22	Deleted		Fig. 3.AN.30	Deleted
3.AM-23	Deleted		3.AO-1	Deleted
3.AM-24	Deleted		3.AP-1	Deleted
3.AM-25	Deleted		3.AQ-1	Deleted
3.AM-26	Deleted		3.AQ-2	Deleted
3.AM-27	Deleted		3.AQ-3	Deleted
3.AM-28	Deleted		3.AQ-4	Deleted
3.AM-29	Deleted		3.AQ-5	Deleted
3.AM-30	Deleted		3.AQ-6	Deleted
3.AN-1	Deleted		3.AQ-7	Deleted
3.AN-2	Deleted		3.AQ-8	Deleted
3.AN-3	Deleted		3.AQ-9	Deleted
3.AN-4	Deleted		3.AQ-10	Deleted
3.AN-5	Deleted		3.AR-1	Deleted
3.AN-6	Deleted		3.AR-2	Deleted
3.AN-7	Deleted		3.AR-3	Deleted
3.AN-8	Deleted		3.AR-4	Deleted
3.AN-9	Deleted		3.AR-5	Deleted
3.AN-10	Deleted		3.AR-6	Deleted
3.AN-11	Deleted		3.AR-7	Deleted
3.AN-12	Deleted		3.AR-8	Deleted
3.AN-13	Deleted		3.AR-9	Deleted
3.AN-14	Deleted		3.AR-10	Deleted
Fig. 3.AN.1	Deleted		3.AR-11	Deleted
Fig. 3.AN.2	Deleted		3.AS-1	Deleted
Fig. 3.AN.3	Deleted		3.AS-2	Deleted
Fig. 3.AN.4	Deleted		3.AS-3	Deleted
Fig. 3.AN.5	Deleted		3.AS-4	Deleted
Fig. 3.AN.6	Deleted		3.AS-5	Deleted
Fig. 3.AN.7	Deleted		3.AS-6	Deleted
Fig. 3.AN.8	Deleted		3.AS-7	Deleted
Fig. 3.AN.9	Deleted		3.AS-8	Deleted
Fig. 3.AN.10	Deleted		3.AS-9	Deleted
Fig. 3.AN.11	Deleted		3.AS-10	Deleted
Fig. 3.AN.12	Deleted		3.AS-11	Deleted
Fig. 3.AN.13	Deleted		3.AS-12	Deleted
Fig. 3.AN.14	Deleted		3.AS-13	Deleted

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
4.0-1	2C		4.3-23	Deleted
4.0-2	2C		Fig. 4.3.1	Deleted
4.0-3	2C		Fig. 4.3.2	Deleted
4.0-4	2C		Fig. 4.3.3	Deleted
4.0-5	2C		Fig. 4.3.4	Deleted
Fig. 4.0.1	2A		4.4-1	2C
Fig. 4.0.2	2B		4.4-2	2C
Fig. 4.0.3	2B		4.4-3	2C
4.1-1	2C		4.4-4	2C
4.1-2	2C		4.4-5	2C
4.1-3	2C		4.4-6	2C
4.1-4	2C		4.4-7	2C
4.1-5	2C		4.4-8	2C
4.1-6	2C		4.4-9	2C
4.1-7	Deleted		4.4-10	2C
4.2-1	2B		4.4-11	2C
4.2-2	2B		4.4-12	2C
4.2-3	2B		4.4-13	2C
4.2-4	2B		4.4-14	2C
4.2-5	2B		4.4-15	2C
4.2-6	2B		4.4-16	2C
4.2-7	2B		4.4-17	2C
4.2-8	2B		4.4-18	2C
4.2-9	2B		4.4-19	2C
4.2-10	2B		4.4-20	2C
4.2-11	2B		4.4-21	2C
4.2-12	2B		4.4-22	2C
Fig. 4.2.1	Deleted		4.4-23	2C
Fig. 4.2.2	Deleted		4.4-24	2C
Fig. 4.2.3	2B		4.4-25	2C
4.3-1	2C		4.4-26	2C
4.3-2	2C		4.4-27	2C
4.3-3	2C		4.4-28	2C
4.3-4	2C		4.4-29	2C
4.3-5	Deleted		4.4-30	2C
4.3-6	Deleted		4.4-31	2C
4.3-7	Deleted		4.4-32	2C
4.3-8	Deleted		4.4-33	2C
4.3-9	Deleted		4.4-34	2C
4.3-10	Deleted		4.4-35	2C
4.3-11	Deleted		4.4-36	2C
4.3-12	Deleted		4.4-37	2C
4.3-13	Deleted		4.4-38	2C
4.3-14	Deleted		4.4-39	2C
4.3-15	Deleted		4.4-40	2C
4.3-16	Deleted		4.4-41	2C
4.3-17	Deleted		4.4-42	2C
4.3-18	Deleted		4.4-43	2C
4.3-19	Deleted		4.4-44	2C
4.3-20	Deleted		4.4-45	2C
4.3-21	Deleted		4.4-46	2C
4.3-22	Deleted		4.4-47	2C

Page	Revision		Page	Revision
4.4-48	2C		Fig. 4.4.31	2B
4.4-49	2C		Fig. 4.4.32	2B
4.4-50	2C		Fig. 4.4.33	2B
4.4-51	Deleted		Fig. 4.4.34	2B
4.4-52	Deleted		Fig. 4.4.35	2B
4.4-53	Deleted		4.5-1	2C
4.4-54	Deleted		4.5-2	2C
4.4-55	Deleted		4.5-3	2C
4.4-56	Deleted		4.5-4	2C
4.4-57	Deleted		4.5-5	2C
4.4-58	Deleted		4.5-6	2C
4.4-59	Deleted		4.5-7	2C
4.4-60	Deleted		4.5-8	2C
4.4-61	Deleted		4.5-9	2C
4.4-62	Deleted		4.5-10	2C
4.4-63	Deleted		4.5-11	2C
4.4-64	Deleted		4.5-12	2C
4.4-65	Deleted		4.5-13	2C
4.4-66	Deleted		4.5-14	2C
4.4-67	Deleted		4.5-15	2C
4.4-68	Deleted		4.5-16	2C
Fig. 4.4.1	0		4.5-17	2C
Fig. 4.4.2	0		4.5-18	2C
Fig. 4.4.3	0		4.5-19	2C
Fig. 4.4.4	0		4.5-20	2C
Fig. 4.4.5	0		4.5-21	2C
Fig. 4.4.6	2A		4.5-22	2C
Fig. 4.4.7	2B		4.5-23	2C
Fig. 4.4.8	Deleted		4.5-24	2C
Fig. 4.4.9	1		4.5-25	Deleted
Fig. 4.4.10	0		4.5-26	Deleted
Fig. 4.4.11	Deleted		Fig. 4.5.1	Deleted
Fig. 4.4.12	Deleted		Fig. 4.5.2	Deleted
Fig. 4.4.13	0		Fig. 4.5.3	Deleted
Fig. 4.4.14	Deleted		Fig. 4.5.4	2B
Fig. 4.4.15	Deleted		4.6-1	2C
Fig. 4.4.16	2B		4.6-2	2C
Fig. 4.4.17	2B		4.7-1	2C
Fig. 4.4.18	Deleted		4.7-2	2C
Fig. 4.4.19	2B		4.7-3	2C
Fig. 4.4.20	2B		4.A-1	Deleted
Fig. 4.4.21	Deleted		4.A-2	Deleted
Fig. 4.4.22	Deleted		4.A-3	Deleted
Fig. 4.4.23	Deleted		4.A-4	Deleted
Fig. 4.4.24	0		4.A-5	Deleted
Fig. 4.4.25	2B		4.A-6	Deleted
Fig. 4.4.26	2B		4.A-7	Deleted
Fig. 4.4.27	2		4.A-8	Deleted
Fig. 4.4.28	2B		4.A-9	Deleted
Fig. 4.4.29	2A		4.A-10	Deleted
Fig. 4.4-30	2B		4.A-11	Deleted

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
4.A-12	Deleted			
4.A-13	Deleted			
4.A-14	Deleted			
4.A-15	Deleted			
4.A-16	Deleted			
4.A-17	Deleted			
4.A-18	Deleted			
4.A-19	Deleted			
4.A-20	Deleted			
4.A-21	Deleted			
4.A-22	Deleted			
4.A-23	Deleted			
Fig. 4.A.1	Deleted			
Fig. 4.A.2	Deleted			
Fig. 4.A.3	Deleted			
Fig. 4.A.4	Deleted			
Fig. 4.A.5	Deleted			
Fig. 4.A.6	Deleted			
Fig. 4.A.7	Deleted			
Fig. 4.A.8	Deleted			
Fig. 4.A.9	Deleted			
Fig. 4.A.10	Deleted			
Fig. 4.A.11	Deleted			
Fig. 4.A.12	Deleted			
Fig. 4.A.13	Deleted			
4.B-1	2C			
4.B-2	2C			
4.B-3	2C			
4.B-4	2C			
4.B-5	2C			
4.B-6	2C			
4.B-7	2C			
4.B-8	2C			
4.B-9	2C			
4.B-10	2C			
4.B-11	2C			
4.B-12	2C			
4.B-13	2C			
Fig. 4.B.1	1			
Fig. 4.B.2	1			
Fig. 4.B.3	1			
Fig. 4.B.4	1			
Fig. 4.B.5	1			
Fig. 4.B.6	1			
Fig. 4.B.7	1			
Fig. 4.B.8	Deleted			

Page	Revision		Page	Revision
5.0-1	2B		5.2-18	2B
5.0-2	2B		5.2-19	2B
5.0-3	2B		5.2-20	2B
5.1-1	2B		5.2-21	2B
5.1-2	2B		5.2-22	2B
5.1-3	2B		5.2-23	2B
5.1-4	2B		5.2-24	2B
5.1-5	2B		5.2-25	2B
5.1-6	2B		5.2-26	2B
5.1-7	2B		5.2-27	2B
5.1-8	2B		5.2-28	2B
5.1-9	2B		5.2-29	2B
5.1-10	2B		5.2-30	2B
5.1-11	2B		5.2-31	2B
5.1-12	2B		5.2-32	2B
5.1-13	2B		5.2-33	2B
5.1-14	2B		5.2-34	2B
5.1-15	2B		5.2-35	2B
5.1-16	2B		5.2-36	2B
5.1-17	2B		5.2-37	2B
5.1-18	2B		5.2-38	2B
5.1-19	2B		5.2-39	2B
5.1-20	2B		5.2-40	2B
Fig. 5.1.1	1		5.2-41	2B
Fig. 5.1.2	0		5.2-42	2B
Fig. 5.1.3	2B		5.2-43	2B
Fig. 5.1.4	0		5.2-44	2B
Fig. 5.1.5	0		5.2-45	2B
Fig. 5.1.6	0		5.2-46	2B
Fig. 5.1.7	0		5.2-47	2B
Fig. 5.1.8	0		5.2-48	2B
Fig. 5.1.9	0		5.2-49	2B
Fig. 5.1.10	0		5.2-50	2B
Fig. 5.1.11	0		5.2-51	2B
Fig. 5.1.12	1		5.2-52	2B
5.2-1	2B		5.2-53	2B
5.2-2	2B		5.2-54	2B
5.2-3	2B		5.2-55	2B
5.2-4	2B		5.2-56	2B
5.2-5	2B		5.2-57	2B
5.2-6	2B		5.2-58	2B
5.2-7	2B		5.3-1	2A
5.2-8	2B		5.3-2	2A
5.2-9	2B		5.3-3	2A
5.2-10	2B		5.3-4	2A
5.2-11	2B		5.3-5	2A
5.2-12	2B		5.3-6	2A
5.2-13	2B		5.3-7	2A
5.2-14	2B		5.3-8	2A
5.2-15	2B		5.3-9	2A
5.2-16	2B		5.3-10	2A
5.2-17	2B		5.3-11	2A

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
5.3-12	2A		5.4-29	2B
5.3-13	2A		5.4-30	2B
5.3-14	2A		5.4-31	2B
Fig. 5.3.1	1		5.4-32	2B
Fig. 5.3.2	0		5.4-33	2B
Fig. 5.3.3	0		5.4-34	2B
Fig. 5.3.4	1		5.4-35	2B
Fig. 5.3.5	0		5.5-1	0
Fig. 5.3.6	0		5.6-1	2B
Fig. 5.3.7	1		5.6-2	2B
Fig. 5.3.8	0		5.6-3	2B
Fig. 5.3.9	0		5.A-1	0
Fig. 5.3.10	1		5.A-2	0
Fig. 5.3.11	1		5.A-3	0
Fig. 5.3.12	0		5.B-1	0
Fig. 5.3.13	0		5.B-2	0
Fig. 5.3.14	1		5.B-3	0
Fig. 5.3.15	1		5.B-4	0
Fig. 5.3.16	1		5.B-5	0
Fig. 5.3.17	1		5.B-6	0
Fig. 5.3.18	1		5.B-7	0
Fig. 5.3.19	1		5.C-1	0
Fig. 5.3-20	1		5.C-2	0
Fig. 5.3-21	1		5.C-3	0
5.4-1	2B		5.C-4	0
5.4-2	2B		5.C-5	0
5.4-3	2B		5.C-6	0
5.4-4	2B		5.C-7	0
5.4-5	2B		5.C-8	0
5.4-6	2B		5.C-9	0
5.4-7	2B		5.C-10	0
5.4-8	2B		5.C-11	0
5.4-9	2B		5.C-12	0
5.4-10	2B		5.C-13	0
5.4-11	2B		5.C-14	0
5.4-12	2B		5.C-15	0
5.4-13	2B		5.C-16	0
5.4-14	2B		5.C-17	0
5.4-15	2B		5.C-18	0
5.4-16	2B		5.C-19	0
5.4-17	2B		5.C-20	0
5.4-18	2B		5.C-21	0
5.4-19	2B		5.C-22	0
5.4-20	2B		5.C-23	0
5.4-21	2B		5.C-24	0
5.4-22	2B		5.C-25	0
5.4-23	2B		5.C-26	0
5.4-24	2B		5.C-27	0
5.4-25	2B		5.C-28	0
5.4-26	2B		5.C-29	0
5.4-27	2B		5.C-30	0
5.4-28	2B		5.C-31	0

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
5.C-32	0			
5.C-33	0			
5.C-34	0			
5.C-35	0			
5.C-36	0			
5.C-37	0			
5.C-38	0			
5.C-39	0			
5.C-40	0			
5.C-41	0			
5.C-42	0			
5.C-43	0			
5.C-44	0			
5.C-45	0			
5.C-46	0			
5.C-47	0			
5.C-48	0			
5.C-49	0			
5.C-50	0			
5.C-51	0			
5.C-52	0			
5.C-53	0			
5.C-54	0			
5.C-55	0			
5.C-56	0			
5.C-57	0			
5.C-58	0			
5.C-59	0			
5.C-60	0			
5.C-61	0			
5.D-1	0			
5.D-2	0			
5.D-3	0			
5.D-4	0			
5.D-5	0			
5.D-6	0			
5.E-1	1			
5.E-2	1			
5.E-3	1			
5.E-4	1			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
6.1-1	2B		6.2-33	2B
6.1-2	2B		6.2-34	2B
6.1-3	2B		6.2-35	2B
6.1-4	2B		6.2-36	2B
6.1-5	2B		6.2-37	2B
6.1-6	2B		6.2-38	2B
6.1-7	2B		6.2-39	2B
6.1-8	2B		6.2-40	2B
6.1-9	2B		6.2-41	2B
6.1-10	2B		6.2-42	2B
6.1-11	2B		6.2-43	2B
6.1-12	2B		6.2-44	2B
6.1-13	2B		6.2-45	2B
6.1-14	2B		6.2-46	2B
6.1-15	2B		6.2-47	2B
6.1-16	2B		6.2-48	2B
6.1-17	2B		6.2-49	2B
6.1-18	2B		6.2-50	2B
6.1-19	2B		6.2-51	2B
6.2-1	2B		6.2-52	2B
6.2-2	2B		6.2-53	2B
6.2-3	2B		6.2-54	2B
6.2-4	2B		6.2-55	2B
6.2-5	2B		6.2-56	2B
6.2-6	2B		6.2-57	2B
6.2-7	2B		6.2-58	2B
6.2-8	2B		6.2-59	2B
6.2-9	2B		6.2-60	2B
6.2-10	2B		6.2-61	2B
6.2-11	2B		6.2-62	2B
6.2-12	2B		6.2-63	2B
6.2-13	2B		6.2-64	2B
6.2-14	2B		6.3-1	2A
6.2-15	2B		6.3-2	2A
6.2-16	2B		6.3-3	2A
6.2-17	2B		6.3-4	2A
6.2-18	2B		6.3-5	2A
6.2-19	2B		6.3-6	2A
6.2-20	2B		6.3-7	2A
6.2-21	2B		6.3-8	2A
6.2-22	2B		6.3-9	2A
6.2-23	2B		6.3-10	2A
6.2-24	2B		6.3-11	2A
6.2-25	2B		6.3-12	2A
6.2-26	2B		6.3-13	2A
6.2-27	2B		6.3-14	2A
6.2-28	2B		6.3-15	2A
6.2-29	2B		6.3-16	2B
6.2-30	2B		6.3-17	2B
6.2-31	2B		6.3-18	2B
6.2-32	2B		Fig. 6.3.1	1

Page	Revision		Page	Revision
Fig. 6.3.1A	1		Fig. 6.4.10	1
Fig. 6.3.2	1		Fig. 6.4.11	1
Fig. 6.3.3	0		Fig. 6.4.12	1
Fig. 6.3.4	1		Fig. 6.4.13	1
Fig. 6.3.4A	1		Fig. 6.4.14	1
Fig. 6.3.5	1		Fig. 6.4.15	1
Fig. 6.3.6	0		Fig. 6.4.16	2
Fig. 6.3.7	1		Fig. 6.4.17	2A
6.4-1	2B		6.5-1	0
6.4-2	2B		6.6-1	0
6.4-3	2B		6.7-1	1
6.4-4	2B		6.7-2	1
6.4-5	2B		6.A-1	1
6.4-6	2B		6.A-2	1
6.4-7	2B		6.A-3	1
6.4-8	2B		6.A-4	1
6.4-9	2B		6.A-5	1
6.4-10	2B		6.A-6	1
6.4-11	2B		6.A-7	1
6.4-12	2B		6.A-8	1
6.4-13	2B		6.A-9	1
6.4-14	2B		6.A-10	1
6.4-15	2B		6.A-11	1
6.4-16	2B		6.A-12	1
6.4-17	2B		6.A-13	1
6.4-18	2B		6.A-14	1
6.4-19	2B		6.A-15	1
6.4-20	2B		6.A-16	1
6.4-21	2B		6.A-17	1
6.4-22	2B		6.A-18	1
6.4-23	2B		6.A-19	1
6.4-24	2B		6.A-20	1
6.4-25	2B		Fig. 6.A.1	0
6.4-26	2B		Fig. 6.A.2	0
6.4-27	2B		Fig. 6.A.3	0
6.4-28	2B		Fig. 6.A.4	0
6.4-29	2B		Fig. 6.A.5	0
6.4-30	2B		Fig. 6.A.6	0
6.4-31	2B		6.B-1	0
6.4-32	2B		6.B-2	0
6.4-33	2B		6.C-1	2B
Fig. 6.4.1	Deleted		6.C-2	2B
Fig. 6.4.2	1		6.C-3	2B
Fig. 6.4.3	1		6.C-4	2B
Fig. 6.4.4	1		6.C-5	2B
Fig. 6.4.5	1		6.C-6	2B
Fig. 6.4.6	1		6.C-7	2B
Fig. 6.4.7	1		6.C-8	2B
Fig. 6.4.8	1		6.C-9	2B
Fig. 6.4.9	1		6.C-10	2B

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
6.C-11	2B			
6.C-12	2B			
6.C-13	2B			
6.C-14	2B			
6.C-15	2B			
6.C-16	2B			
6.C-17	2B			
6.D-1	1			
6.D-2	1			
6.D-3	1			
6.D-4	1			
6.D-5	1			
6.D-6	1			
6.D-7	1			
6.D-8	1			
6.D-9	1			
6.D-10	1			
6.D-11	1			
6.D-12	1			
6.D-13	1			
6.D-14	1			
6.D-15	1			
6.D-16	1			
6.D-17	1			
6.D-18	1			
6.D-19	1			
6.D-20	1			
6.D-21	1			
6.D-22	1			
6.D-23	1			
6.D-24	1			
6.D-25	1			
6.D-26	1			
6.D-27	1			
6.D-28	1			
6.D-29	1			
6.D-30	1			
6.D-31	1			
6.D-32	1			
6.D-33	1			
6.D-34	1			
6.D-35	1			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
7.0-1	2B		7.A-8	Deleted
7.1-1	2B		7.A-9	Deleted
7.1-2	2B		7.A-10	Deleted
7.1-3	2B		7.A-11	Deleted
7.1-4	2B		7.A-12	Deleted
7.1-5	2B		7.A-13	Deleted
7.1-6	2B		7.A-14	Deleted
7.1-7	2B		7.A-15	Deleted
7.1-8	2B		7.A-16	Deleted
Fig. 7.1.1	0		7.A-17	Deleted
7.2-1	2B		7.A-18	Deleted
7.2-2	Deleted		7.A-19	Deleted
7.2-3	Deleted		7.A-20	Deleted
7.2-4	Deleted		7.A-21	Deleted
7.2-5	Deleted		7.A-22	Deleted
7.2-6	Deleted		7.A-23	Deleted
7.2-7	Deleted		7.A-24	Deleted
7.2-8	Deleted		7.A-25	Deleted
7.2-9	Deleted		7.A-26	Deleted
7.2-10	Deleted		7.A-27	Deleted
7.2-11	Deleted		7.A-28	Deleted
7.2-12	Deleted		7.A-29	Deleted
7.3-1	2B		7.A-30	Deleted
7.3-2	Deleted		7.A-31	Deleted
7.3-3	Deleted		7.A-32	Deleted
7.3-4	Deleted		7.A-33	Deleted
7.3-5	Deleted		7.A-34	Deleted
7.3-6	Deleted		7.A-35	Deleted
7.3-7	Deleted		7.A-36	Deleted
7.3-8	Deleted		7.A-37	Deleted
7.3-9	Deleted		7.A-38	Deleted
7.3-10	Deleted		7.A-39	Deleted
7.3-11	Deleted		7.A-40	Deleted
7.3-12	Deleted		7.A-41	Deleted
7.3-13	Deleted		7.A-42	Deleted
7.3-14	Deleted		7.A-43	Deleted
7.3-15	Deleted		7.A-44	Deleted
7.3-16	Deleted		7.A-45	Deleted
7.3-17	Deleted		7.A-46	Deleted
7.3-18	Deleted		7.A-47	Deleted
7.3-19	Deleted		7.A-48	Deleted
7.3-20	Deleted		7.A-49	Deleted
7.3-21	Deleted		7.A-50	Deleted
7.4-1	2B		7.A-51	Deleted
7.4-2	2B		7.A-52	Deleted
7.A-1	2B		7.A-53	Deleted
7.A-2	Deleted		7.A-54	Deleted
7.A-3	Deleted		7.A-55	Deleted
7.A-4	Deleted		7.A-56	Deleted
7.A-5	Deleted		7.A-57	Deleted
7.A-6	Deleted		7.A-58	Deleted
7.A-7	Deleted		7.A-59	Deleted

Page	Revision		Page	Revision
7.A-60	Deleted			
7.A-61	Deleted			
7.A-62	Deleted			
7.A-63	Deleted			
7.A-64	Deleted			
7.A-65	Deleted			
7.A-66	Deleted			
7.A-67	Deleted			
7.A-68	Deleted			
7.A-69	Deleted			
7.A-70	Deleted			
7.A-71	Deleted			
7.A-72	Deleted			
7.A-73	Deleted			
7.A-74	Deleted			
7.A-75	Deleted			
7.A-76	Deleted			
7.A-77	Deleted			
7.A-78	Deleted			
7.A-79	Deleted			
7.A-80	Deleted			
7.A-81	Deleted			
7.A-82	Deleted			
7.A-83	Deleted			
7.A-84	Deleted			
7.A-85	Deleted			
7.A-86	Deleted			
7.A-87	Deleted			
7.A-88	Deleted			
7.A-89	Deleted			
7.A-90	Deleted			
7.A-91	Deleted			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
8.0-1	2B		Fig. 8.1.2d	1
8.0-2	2B		Fig. 8.1.2e	0
8.0-3	2B		Fig. 8.1.2f	0
8.0-4	2B		Fig. 8.1.3	0
8.0-5	2B		Fig. 8.1.4	1
8.0-6	2B		Fig. 8.1.5	0
8.1-1	2C		Fig. 8.1.6	0
8.1-2	2C		Fig. 8.1.7	1
8.1-3	2C		Fig. 8.1.8	0
8.1-4	2C		Fig. 8.1.9	0
8.1-5	2C		Fig. 8.1.10	0
8.1-6	2C		Fig. 8.1.11	0
8.1-7	2C		Fig. 8.1.12	1
8.1-8	2C		Fig. 8.1.13	1
8.1-9	2C		Fig. 8.1.14	0
8.1-10	2C		Fig. 8.1.15	0
8.1-11	2C		Fig. 8.1.16	0
8.1-12	2C		Fig. 8.1.17	0
8.1-13	2C		Fig. 8.1.18	1
8.1-14	2C		Fig. 8.1.19	1
8.1-15	2C		Fig. 8.1.20	1
8.1-16	2C		Fig. 8.1.21	1
8.1-17	2C		Fig. 8.1.22a	1
8.1-18	2C		Fig. 8.1.22b	1
8.1-19	2C		Fig. 8.1.23	1
8.1-20	2C		Fig. 8.1.24	0
8.1-21	2C		Fig. 8.1.25	1
8.1-22	2C		Fig. 8.1.26	1
8.1-23	2C		Fig. 8.1.27	1
8.1-24	2C		Fig. 8.1.28	1
8.1-25	2C		Fig. 8.1.29a	1
8.1-26	2C		Fig. 8.1.29b	1
8.1-27	2C		Fig. 8.1.30	1
8.1-28	2C		Fig. 8.1.31	1
8.1-29	2C		Fig. 8.1.32	0
8.1-30	2C		Fig. 8.1.33	1
8.1-31	2C		Fig. 8.1.34a	1
8.1-32	2C		Fig. 8.1.34b	Deleted
8.1-33	2C		Fig. 8.1.35	0
8.1-34	2C		Fig. 8.1.36	0
8.1-35	2C		Fig. 8.1.37	1
8.1-36	2C		8.2-1	1
8.1-37	2C		8.3-1	2C
8.1-38	2C		8.3-2	2C
8.1-39	2C		8.3-3	2C
8.1-40	2C		8.3-4	2C
8.1-41	2C		8.3-5	2C
8.1-42	2C		8.3-6	2C
Fig. 8.1.1	1		8.3-7	2C
Fig. 8.1.2a	0		8.3-8	2C
Fig. 8.1.2b	0		8.3-9	2C
Fig. 8.1.2c	1		8.3-10	2C

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
8.3-11	2C			
Fig. 8.3.1	1			
Fig. 8.3.2a	1			
Fig. 8.3.2b	0			
Fig. 8.3.2c	0			
Fig. 8.3.2d	0			
Fig. 8.3.3	0			
Fig. 8.3.4	1			
8.4-1	2A			
8.4-2	2A			
Fig. 8.4-1a	1			
Fig. 8.4-1b	1			
8.5-1	2A			
8.5-2	2A			
8.5-3	2A			
8.5-4	2A			
8.5-5	2A			
Fig. 8.5-1	0			
8.6-1	0			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
9.0-1	0			
9.1-1	2C			
9.1-2	2C			
9.1-3	2C			
9.1-4	2C			
9.1-5	2C			
9.1-6	2C			
9.1-7	2C			
9.1-8	2C			
9.1-9	2C			
9.1-10	2C			
9.1-11	2C			
9.1-12	2C			
9.1-13	2C			
9.1-14	2C			
9.1-15	2C			
9.1-16	2C			
9.1-17	2B			
9.1-18	2B			
9.1-19	2B			
9.1-20	2B			
9.1-21	2B			
9.1-22	2B			
9.1-23	2B			
9.1-24	2B			
9.1-25	2B			
9.1-26	2B			
9.1-27	2B			
9.1-28	2B			
9.1-29	2B			
9.1-30	2B			
9.2-1	2B			
9.2-2	2B			
9.2-3	2B			
9.2-4	2B			
9.3-1	2B			
9.4-1	0			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
10.1-1	1		10.3-42	2A
10.1-2	1		10.3-43	2A
10.1-3	1		Fig. 10.3.1a	1
10.1-4	1		Fig. 10.3.1b	1
10.1-5	1		Fig. 10.3.1c	1
10.1-6	1		Fig. 10.3.1d	1
10.1-7	1		Fig. 10.3.1e	1
10.1-8	1		10.4-1	2B
Fig. 10.1.1	1		10.4-2	2B
10.2-1	2A		10.4-3	2B
10.3-1	2A		10.4-4	2B
10.3-2	2A		10.4-5	2B
10.3-3	2C		10.5-1	0
10.3-4	2C			
10.3-5	2B			
10.3-6	2B			
10.3-7	2B			
10.3-8	2B			
10.3-9	2B			
10.3-10	2B			
10.3-11	2B			
10.3-12	2B			
10.3-13	2B			
10.3-14	2B			
10.3-15	2B			
10.3-16	2B			
10.3-17	2B			
10.3-18	2B			
10.3-19	2B			
10.3-20	2B			
10.3-21	2B			
10.3-22	2A			
10.3-23	2A			
10.3-24	2A			
10.3-25	2A			
10.3-26	2A			
10.3-27	2A			
10.3-28	2A			
10.3-29	2A			
10.3-30	2A			
10.3-31	2A			
10.3-32	2A			
10.3-33	2A			
10.3-34	2A			
10.3-35	2A			
10.3-36	2A			
10.3-37	2A			
10.3-38	2A			
10.3-39	2A			
10.3-40	2A			
10.3-41	2A			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
11.1-1	2C		11.2-33	2B
11.1-2	2C		11.2-34	2B
11.1-3	2C		11.2-35	2B
11.1-4	2C		11.2-36	2B
11.1-5	2C		11.2-37	2B
11.1-6	2C		11.2-38	2B
11.1-7	2C		11.2-39	2B
11.1-8	2C		11.2-40	2B
11.1-9	2C		11.2-41	2B
11.1-10	2C		11.2-42	2B
11.1-11	2C		11.2-43	2B
11.1-12	2C		11.2-44	2B
11.1-13	2C		11.2-45	2B
11.1-14	2C		11.2-46	2B
11.1-15	2C		11.2-47	2B
11.1-16	2C		11.2-48	2B
11.1-17	2C		11.2-49	2B
11.1-18	2C		11.2-50	2B
11.1-19	2C		11.2-51	2B
11.1-20	2C		Fig. 11.2.1	0
11.2-1	2B		Fig. 11.2.2	Deleted
11.2-2	2B		Fig. 11.2.3	Deleted
11.2-3	2B		Fig. 11.2.4	Deleted
11.2-4	2B		Fig. 11.2.5	Deleted
11.2-5	2B		Fig. 11.2.6	2A
11.2-6	2B		Fig. 11.2.7	Deleted
11.2-7	2B		Fig. 11.2.8	2C
11.2-8	2B		11.3-1	1
11.2-9	2B			
11.2-10	2B			
11.2-11	2B			
11.2-12	2B			
11.2-13	2B			
11.2-14	2B			
11.2-15	2B			
11.2-16	2B			
11.2-17	2B			
11.2-18	2B			
11.2-19	2B			
11.2-20	2B			
11.2-21	2B			
11.2-22	2B			
11.2-23	2B			
11.2-24	2B			
11.2-25	2B			
11.2-26	2B			
11.2-27	2B			
11.2-28	2B			
11.2-29	2B			
11.2-30	2B			
11.2-31	2B			
11.2-32	2B			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
12.0-1	0		B 3.1.4-3	2C
12.1-1	2C		B 3.1.4-4	2C
12.1-2	2C		B 3.2.1-1	2A
12.1-3	2C		B 3.2.1-2	Deleted
12.2-1	2C		B 3.2.1-3	Deleted
12.2-2	2C		B 3.2.2-1	2A
12.2-3	2C		B 3.2.2-2	Deleted
12.2-4	2C		B 3.2.2-3	Deleted
12.2-5	2C		B 3.2.3-1	2A
12.2-6	2C		B 3.2.3-2	Deleted
12.2-7	2C		B 3.2.3-3	Deleted
12.2-8	2C		B 3.3.1-1	2A
12.2-9	2C		B 3.3.1-2	2A
12.3-1	0		B 3.3.1-3	2A
12.4-1	0		B 3.3.1-4	2A
12.5-1	0		B 3.3.1-5	2A
Appendix 12.A Cover	2B		Appendix 12.B Cover	0
TS Bases TOC	2C		Comment Resolution Letters	0
B 3.0-1	0		21 Pages	
B 3.0-2	0			
B 3.0-3	0			
B 3.0-4	0			
B 3.0-5	0			
B 3.0-6	0			
B 3.0-7	0			
B 3.0-8	0			
B 3.0-9	0			
B 3.1.1-1	2B			
B 3.1.1-2	2B			
B 3.1.1-3	2B			
B 3.1.1-4	2B			
B 3.1.1-5	2B			
B 3.1.1-6	2B			
B 3.1.1-7	2B			
B 3.1.1-8	2B			
B 3.1.2-1	2C			
B 3.1.2-2	2C			
B 3.1.2-3	2C			
B 3.1.2-4	2C			
B 3.1.2-5	2C			
B 3.1.2-6	2C			
B 3.1.2-7	2C			
B 3.1.3-1	2B			
B 3.1.3-2	2B			
B 3.1.3-3	2B			
B 3.1.3-4	2B			
B 3.1.3-5	2B			
B 3.1.3-6	2B			
B 3.1.3-7	2B			
B 3.1.4-1	2C			
B 3.1.4-2	2C			

LIST OF EFFECTIVE PAGES FOR HI-STORM 100 FSAR PROPOSED REVISION 2C

Page	Revision		Page	Revision
13.0-1	2B			
13.0-2	2B			
13.1-1	Deleted			
13.1-2	Deleted			
13.2-1	Deleted			
13.3-1	Deleted			
13.3-2	Deleted			
13.3-3	Deleted			
13.3-4	Deleted			
13.3-5	Deleted			
13.3-6	Deleted			
13.3-7	Deleted			
13.3-8	Deleted			
13.3-9	Deleted			
13.3-10	Deleted			
13.3-11	Deleted			
13.3-12	Deleted			
13.3-13	Deleted			
13.3-14	Deleted			
13.3-15	Deleted			
13.3-16	Deleted			
13.3-17	Deleted			
13.4-1	Deleted in Rev. 1			
13.5-1	Deleted			
13.5-2	Deleted			
13.6-1	2A			
Appendix 13.A	Deleted in Rev. 1			
Appendix 13.B	Deleted in Rev. 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
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(1)
	14.III.4 License Termination	10CFR72.54	(1)
3. Structural Evaluation			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.24(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

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
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
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a)(4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	–	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria *ISG-11, Revision* 3	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria *ISG-11, Revision* 3	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	–	–
5. Shielding Evaluation			
5.1 Discussion and Results	–	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	–	5.2
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 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
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..."	Clarification: 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.	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.
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..." 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". 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".	Exception: 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.	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. To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. *The temperature limits for normal conditions are per Paragraph A.4.3 of Appendix A to ACI 349 and temperature limits for*

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
		*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.~~ Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.
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&PV Code."	Clarification: The HI-STORM overpack steel structure is designed in accordance with the ASME B&PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.	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.
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." 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." 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."	Clarification: As described in Section 4.3, all fuel array types authorized for storage *are assigned a single peak fuel cladding temperature limit.* ~~have been evaluated for the peak fuel cladding temperature limit.~~	As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak *normal* fuel cladding temperature limit *of 400°C.* ~~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.~~

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.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."		
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."	Exception: ~~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,~~ n*N*o additional test program is proposed.	*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.~~
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."	Clarification: 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)	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.

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."~~	~~Clarification: 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."	Exception: 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."	Clarification: 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."	Exception: 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
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.	Exception: All free volume calculations use nominal ~~confinement boundary~~ dimensions *with the results reduced by 5% to account for thermal expansion.*~~, but the volume occupied by the MPC internals (i.e., fuel assemblies, fuel basket, etc.) are calculated using maximum weights and minimum densities.~~	~~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~~ *The use of a 5% volume reduction* underpredicts the remaining free volume.
~~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."~~ *7.V.4 "Confinement Analysis. Review the applicant's confinement analysis and the resulting annual dose at the controlled area boundary."*	Exception: ~~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 ($5x10^{-6}$ atm-cm^3/s) plus 50% for conservatism, which yields $7.5x10^{-6}$ atm-cm^3/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.*	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., ~~helium leakage, hydrostatic, and volumetric weld inspection~~*non-destructive examination, pressure testing, and fabrication shop leakage testing*). *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.*
9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."	Clarification: 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.	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.

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
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."	Exception: 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.	The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation. The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures. The composition of the Holtite-A is confirmed by inspection and tests prior to first use. 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.
13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."	*Exception: Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.*	*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.*

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

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 since they were removed from the thermal model in Amendment 2. MPCs both with and without aluminum heat conduction elements installed are compatible with all HI-STORM overpacks. If used,* ~~T~~ *t*hese 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.

1.2.1.3.1 ~~Boral~~ *Fixed* Neutron Absorbers

1.2.1.3.1.1 Boral™

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}B loading requirement. Coupons extracted from production runs were tested using the wet chemistry procedure. The actual ^{10}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}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 B_4C 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®

METAMIC® 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® is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy reinforced with Type 1 ASTM C-750 boron carbide. METAMIC® 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.†, the high performance and reliability of METAMIC® 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® sheets used in the MPCs, the extruded form is rolled down into the required thickness.

METAMIC® 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® 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® ensures that its density is essentially equal to the theoretical density.*
- *The physical and neutronic properties of METAMIC® 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[†] and near-perfect homogeneity.

An evaluation of the manufacturing technology underlying METAMIC® as disclosed in the above-referenced patents and of the extensive third-party tests carried out under the auspices of EPRI makes METAMIC® an acceptable neutron absorber material for use in the MPCs. Holtec's technical position on METAMIC® 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® 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.

---

† 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® 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® 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® are summarized in Subsection 9.1.5.3.

Because of the absence of interconnected porosities, the time required to dehydrate a METAMIC®-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 ^{10}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_4C particles, such as BORAL with an average particle size in excess of 100 microns. METAMIC®, 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®. Further, EPRI's neutron attenuation measurements on 31 and 15 B_4C weight percent METAMIC® showed that METAMIC® exhibits very uniform ^{10}B areal density. This makes it easy to reliably establish and verify the presence and microscopic and macroscopic uniformity of the ^{10}B in the material. Therefore, 90% credit is applied to the minimum ^{10}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 ^{10}B areal density. In Chapter 9 the qualification and on production tests for METAMIC® to support 90% ^{10}B credit are specified. With 90% credit, the target weight percent of boron carbide in METAMIC® 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® is a solid material, there is no capillary path through which spent fuel pool water can penetrate METAMIC® 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® 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® -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® during these operations..

Mechanical properties of 31 wt.% METAMIC® 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].

Mechanical Properties of 31wt.% B_4C METAMIC		
Property	*As-Fabricated*	*After 48 hours of 900°F Temperature Soak*
Yield Strength (psi)	*32937 ±3132*	*28744 ±3246*
Ultimate Strength (psi)	*40141 ±1860*	*34608 ±1513*
Elongation (%)	*1.8 ±0.8*	*5.7 ±3.1*

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_4C METAMIC® indicated by the above table means that METAMIC® 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_4C METAMIC® served as the principal basis for a recent USNRC SER for 31wt.% B_4C METAMIC for used in wet storage [1.2.12]. Additional studies on METAMIC® [1.2.13], EPRI's and others work provide the confidence that 31wt.% B_4C METAMIC® will perform its intended function in the MPCs.

1.2.1.3.1.3 Locational Fixity of Neutron Absorbers

Both Boral and METAMIC® 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;

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.

Based on the MPC decay heat load and the orientation of the transfer cask during onsite transportation, the Supplemental Cooling System (SCS) may be required to be operational during this time period 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 heat load and transfer cask orientation, 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}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).

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†

[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.

† 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, 13th 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® 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.

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^3 (HI-STORM 100 up to Serial Number (S/N) 7), 155 lb/ft^3 (HI-STORM 100 S/N 8 and higher, and HI-STORM 100S)
Density in lid and pedestal (Minimum)	146 lb/ft^3
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
~~Maximum~~ ~~Local~~ *Through-Thickness Section Average*[†] Temperature Limit Under Long Term Conditions	3~~2~~00°F (See Note 3)
~~Maximum~~ *Through-Thickness* Section Average[†] Temperature Limit Under Short Term Conditions	350°F (Appendix A, ~~Subsection~~ *Paragraph* A.4.2)
Aggregate Maximum Value[††] 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)

[†] *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.*

[††] 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†: 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

† 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 <u>all</u> 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 and for relatively high heat load MPCs containing all MBF, 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. Threshold heat loads, below which a loaded MPC may reside in a HI-TRAC transfer cask without supplemental cooling have been established to ensure the fuel cladding temperature limits are met for this normal short-term operating condition. These limits are based on the heat load of the contained MPC and the orientation in which the HI-TRAC is handled. For heat loads higher than the threshold values, 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 $k_{eff} <$ 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®-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* ~~S~~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 Co~~*Canalysis 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~~*to* 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~~ *Based on the heat load of the contained MPC and the orientation in which the transfer cask is handled, the Supplemental Cooling System (SCS) may 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.

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Canister Drying	*≤ 3 torr for ≥ 30 minutes (VDS)* *≤ 21°F exiting the demoisturizer for ≥ 30 minutes or a dew point of the MPC exit gas ≤ 22.9°F for ≥ 30 minutes(FHD)*	*NUREG-1536, ISG-11, Rev. 3*	*Section 4.5, Appendix 2.B*
Canister Backfill Gas	Helium	-	Section ~~12.3.3~~*4.4*
Canister Backfill	*Varies (see Table 1.2.2)*	Thermal Analysis	Section 4.~~3~~*4*
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*	*Section 4.5*
~~Short-Term Allowable~~ Fuel ~~C~~cladding ~~T~~temperature *limit for Off-Normal and Accident Events*	1058° F *(570 °C)*	~~PNL-4835~~*ISG-11, Rev. 3*	Sections 2.0.1 and 4.3
Insolation	Protected by overpack or HI-TRAC	-	Section 4.3
Confinement:		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
NDE:			
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
Leak Testing:			
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	$5x10^{-6}$ atm-cm^3/sec (helium)	-	*Section 9.1*~~Chapter 12 (TS)~~
Monitoring System	None	10CFR72.128(a)(1)	Section 2.3.2.1
~~Hydrostatic~~*Pressure* Testing:			
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
Retrievability:			
Normal and Off-normal:	No Encroachment on Fuel	10CFR72.122(f),(h)(1), & (l)	Sections 3.4, 3.5, and 3.1.2

Table 2.0.2
HI-STORM OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Section 2.0.2
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
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[†]:			
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

[†] 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)	
Thermal:			
Maximum Design Temperatures:			
Concrete			
~~Local Maximum~~ *Through-Thickness Section Average* (Normal)	2300° F	ACI 349 Appendix A *(Paragraph A.4.3)*	*Section 2.0.2, and* Tables *1.D.1 and* 2.2.3
~~Local Maximum~~ *Through-Thickness Section Average* (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 *(other than lid bottom plate)* *Lid Bottom Plate*	3450° F *400°F*	ASME Code Section II, Part D	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	Section 4.4.1.1.8
Confinement:	None	10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	N/A
Retrievability:			
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
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1
Radiation Protection/Shielding:		10CFR72.126 & 10CFR72.128(a)(2)	
Overpack (Normal/Off-normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10

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

Type	Criteria	Basis	FSAR Reference
Beyond Controlled Area During Normal Operation and Anticipated Occurrences	25 mrem/yr. to whole body 75 mrem/yr. to thyroid 25 mrem/yr. To any critical organ	10CFR72.104	Sections 5.1.1, 7.2, and 10.1
At Controlled Area Boundary from Design Basis Accident	5 rem TEDE or sum of DDE and CDE to any individual organ or tissue (other than lens of eye) ≤ 50 rem. 15 rem lens dose. 50 rem shallow dose to skin or extremity.	10CFR72.106	Sections 5.1.2, 7.3, and 10.1
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1.9
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†:		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	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:			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 ~~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

† 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^2	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
Off-Normal Design Event Conditions:		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
Design-Basis (Postulated) Accident Design Events and Conditions:		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
Design-Basis Natural Phenomenon Design Events and Conditions:		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

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[1]. 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

The allowable maximum decay heat per fuel storage location for ZR-clad fuel in regionalized fuel loading shall be calculated as follows. The fuel storage regions are defined in Table 2.1.13. The number of fuel storage locations in each region and the maximum total decay heat per MPC model is provided in Table 2.1.27.

---

[1] *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*

(i) *Choose a value of X between 1 and 6, where X is the ratio of the maximum decay heat per fuel storage location permitted in Region 1 ($q_{Region\ 1}$) to the maximum decay heat permitted per fuel storage location in Region 2 ($q_{Region\ 2}$).*

(ii) *Calculate $q_{Region\ 2}$ using the following equation:*

$$q_{Region\ 2} = (2 \times Q)/[(1 + X^{0.15}) \times (N_{Region\ 1} \times X + N_{Region\ 2})] \qquad \text{Equation 2.1.9.1}$$

Where:

$q_{Region\ 2}$ = *Maximum allowable decay heat per fuel storage location in Region 2 (kW)*

Q = *Maximum allowable heat load for the MPC from Table 2.1.27 (kW)*

X = *Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step (i)*

$N_{Region\ 1}$ = *Number of fuel storage locations in Region 1 from Table 2.1.27*

$N_{Region\ 2}$ = *Number of fuel storage locations in Region 2 from Table 2.1.27*

(iii) *Calculate $q_{Region\ 1}$ using the following equation:*

$$q_{Region\ 1} = X \times q_{Region\ 2} \qquad \text{Equation 2.1.9.2}$$

Where:

$q_{Region\ 1}$ = *Maximum allowable decay heat per fuel storage location in Region 1 (kW)*

X = *Ratio of $q_{Region\ 1}$ to $q_{Region\ 2}$ chosen in Step (i)*

$q_{Region\ 2}$ = *Maximum allowable decay heat per fuel storage location in Region 2 calculated in Step (ii) (kW)*

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*
- *Maximum allowable fuel assembly decay heat*
- *Minimum fuel assembly average enrichment*

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

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

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

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

Equation 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}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 <u>Other Considerations</u>

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 2.1.16

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

Fuel Assembly Array/Class	All Intact Fuel Assemblies		One or More Damaged Fuel Assemblies or Fuel Debris	
	Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)	Initial Enrichment ≤ 5.0 wt.% ^{235}U (ppmb)	Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)	Initial Enrichment ≤ 5.0 wt.% ^{235}U (ppmb)
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G	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

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

PARAMETER	*VALUE*
Fuel Type	*Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class*
Cladding Type	*ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class*
Maximum Initial Enrichment per Assembly	*As specified in Table 2.1.3 for the applicable array/class*
Post-irradiation Cooling Time and Average Burnup per Assembly	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 8 years and ≤ 40,000 MWD/MTU*
Decay Heat Per Fuel Storage Location	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 710 Watts*
Non-Fuel Hardware Burnup and Cooling Time	*As specified in Table 2.1.25*
Fuel Assembly Length	*≤ 176.8 in. (nominal design)*
Fuel Assembly Width	*≤ 8.54 in. (nominal design)*
Fuel Assembly Weight	*≤ 1,680 lbs (including non-fuel hardware)*
Other Limitations	▪ *Quantity is limited to up to 24 PWR intact fuel assemblies.* ▪ *Neutron sources, damaged fuel assemblies and fuel debris are not permitted for storage in MPC-24.* ▪ *BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.* ▪ *CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16* ▪ *Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.*

Table 2.1.18

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	*VALUE (Note 1)*			
Fuel Type(s)	*Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels*	*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)*	*Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without channels*	*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)*
Cladding Type	*ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class*	*ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class*	*ZR*	*ZR*
Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment	*As specified in Table 2.1.4 for the applicable array/class*	*Planar Average:* *$\leq$ 2.7 wt% ^{235}U for array/classes 6x6A, 6x6C, 7x7A, and 8x8A;* *$\leq$ 4.0 wt% ^{235}U for all other array/classes* *Rod:* *As specified in Table 2.1.4*	*As specified in Table 2.1.4 for array/class 6x6B*	*As specified in Table 2.1.4 for array/class 6x6B*

Table 2.1.18 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	*VALUE (Note 1)*			
Post-irradiation Cooling Time and Average Burnup per Assembly	*ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.* *SS clad: Note 4*	*ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.* *SS clad: Note 4.*	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.*	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.*
Decay Heat Per Fuel Storage Location	*ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.* *SS clad: ≤ 95 Watts*	*ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3.* *SS clad: ≤ 95 Watts*	*≤ 115 Watts*	*≤ 115 Watts*
Fuel Assembly Length	*≤ 176.5 in. (nominal design)*	*Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design)* *All Other array/classes: ≤ 176.5 in. (nominal design)*	*≤ 135.0 in. (nominal design)*	*≤ 135.0 in. (nominal design)*
Fuel Assembly Width	*≤ 5.85 in. (nominal design)*	*Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design)* *All Other array/classes: ≤ 5.85 in. (nominal design)*	*≤ 4.70 in. (nominal design)*	*≤ 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	*≤ 700 lbs. (including channels)*	*Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC)* *All Other array/classes: ≤ 700 lbs. (including channels and DFC)*	*≤ 400 lbs, including channels*	*≤ 550 lbs, including channels and DFC*
Other Limitations	▪ *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.* ▪ *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* ▪ *SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50.* ▪ *Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.* ▪ *Fuel debris is not permitted for storage in MPC-68.*			

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 ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a maximum decay heat ≤ 115 Watts.*

3. *Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a maximum decay ≤ 183.5 Watts.*

4. *SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.*

Table 2.1.19

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	*VALUE (Notes 1 and 2)*			
Fuel Type(s)	*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*	*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)*	*Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels*	*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))*
Cladding Type	*ZR*	*ZR*	*ZR*	*ZR*
Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment	*As specified in Table 2.1.4 for the applicable array/class*	*As specified in Table 2.1.4 for the applicable array/class*	*As specified in Table 2.1.4 for array/class 6x6B*	*As specified in Table 2.1.4 for array/class 6x6B*
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.*	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.*	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.*	*Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.*
Decay Heat Per Fuel Storage Location	*≤ 115 Watts*	*≤ 115 Watts*	*≤ 115 Watts*	*≤ 115 Watts*
Fuel Assembly Length	*≤ 135.0 in. (nominal design)*	*≤ 135.0 in. (nominal design)*	*≤ 135.0 in. (nominal design)*	*≤ 135.0 in. (nominal design)*
Fuel Assembly Width	*≤ 4.70 in. (nominal design)*	*≤ 4.70 in. (nominal design)*	*≤ 4.70 in. (nominal design)*	*≤ 4.70 in. (nominal design)*
Fuel Assembly Weight	*≤ 400 lbs, (including channels)*	*≤ 550 lbs, (including channels and DFC)*	*≤ 400 lbs, (including channels)*	*≤ 550 lbs, (including channels and DFC)*

Table 2.1.19 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	*VALUE*
Other Limitations	▪ *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:* *- uranium oxide BWR intact fuel assemblies* *- MOX BWR intact fuel assemblies* *- uranium oxide BWR damaged fuel assemblies in DFCs* *- MOX BWR damaged fuel assemblies in DFCs* *- up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12.* ▪ *Stainless steel channels are not permitted.* ▪ *Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.*

Notes:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*
2. *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

PARAMETER	*VALUE (Note 1)*	
Fuel Type	*Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class*	*Uranium oxide PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)*
Cladding Type	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*
Maximum Initial Enrichment per Assembly	*As specified in Table 2.1.3 for the applicable array/class*	*As specified in Table 2.1.3 for the applicable array/class*
Post-irradiation Cooling Time, and Average Burnup per Assembly	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU*
Decay Heat Per Fuel Storage Location	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 710 Watts*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 710 Watts*
Non-fuel hardware post-irradiation Cooling Time and Burnup	*As specified in Table 2.1.25*	*As specified in Table 2.1.25*
Fuel Assembly Length	*≤ 176.8 in. (nominal design)*	*≤ 176.8 in. (nominal design)*
Fuel Assembly Width	*≤ 8.54 in. (nominal design)*	*≤ 8.54 in. (nominal design)*
Fuel Assembly Weight	*≤ 1680 lbs (including non-fuel hardware)*	*≤ 1680 lbs (including DFC and non-fuel hardware)*

Table 2.1.20 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E

PARAMETER	*VALUE*
Other Limitations	▪ *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.* ▪ *Fuel debris and neutron sources are not authorized for storage in the MPC-24E.* ▪ *BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.* ▪ *CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16.* ▪ *Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.*

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

PARAMETER	*VALUE (Note 1)*	
Fuel Type	*Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.*	*Uranium oxide, PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.*
Cladding Type	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*
Maximum Initial Enrichment per Assembly	*As specified in Table 2.1.3 for the applicable fuel assembly array/class*	*As specified in Table 2.1.3 for the applicable fuel assembly array/class*
Post-irradiation Cooling Time and Average Burnup per Assembly	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000 MWD/MTU*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000 MWD/MTU*
Decay Heat Per Fuel Storage Location	*ZR-clad: As specified in Section 2.1.9.1* *SS-clad: ≤ 500 Watts*	*ZR-clad: As specified in Section 2.1.9.1* *SS-clad: ≤ 500 Watts*
Non-fuel hardware post-irradiation cooling time and burnup	*As specified in Table 2.1.25*	*As specified in Table 2.1.25*
Fuel Assembly Length	*≤ 176.8 in. (nominal design)*	*≤ 176.8 in. (nominal design)*
Fuel Assembly Width	*≤ 8.54 in. (nominal design)*	*≤ 8.54 in. (nominal design)*
Fuel Assembly Weight	*≤ 1,680 lbs (including non-fuel hardware)*	*≤ 1,680 lbs (including DFC and non-fuel hardware)*

Table 2.1.21 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32

PARAMETER	*VALUE*
Other Limits	▪ *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.* ▪ *Fuel debris and neutron sources are not permitted for storage in MPC-32.* ▪ *BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.* ▪ *CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20.* ▪ *Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.*

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

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

Table 2.1.22 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68FF

PARAMETER	*VALUE (Note 1)*	
Fuel Assembly Width	*Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design)* *All Other array/classes:* *≤ 5.85 in. (nominal design)*	*Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design)* *All Other array/classes:* *≤ 5.85 in. (nominal design)*
Fuel Assembly Weight	*Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels)* *All Other array/classes:* *≤ 700 lbs. (including channels)*	*Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC)* *All Other array/classes:* *≤ 700 lbs. (including channels and DFC)*
Other Limitations	▪ *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.* ▪ *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.* ▪ *SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50.* ▪ *Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.*	

NOTES:

1. *A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.*

2. *Array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a maximum decay heat ≤ 115 Watts.*

3. *Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a maximum decay ≤ 183.5 Watts.*

4. *SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.*

Table 2.1.23

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

PARAMETER	*VALUE (Note 1)*	
Fuel Type	*Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class*	*Uranium oxide PWR damaged fuel assemblies and/or fuel debris meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)*
Cladding Type	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*	*ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class*
Maximum Initial Enrichment per Assembly	*As specified in Table 2.1.3 for the applicable array/class*	*As specified in Table 2.1.3 for the applicable array/class*
Post-irradiation Cooling Time, and Average Burnup per Assembly	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU*
Decay Heat Per Fuel Storage Location	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 710 Watts*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 710 Watts*
Non-fuel hardware post-irradiation Cooling Time and Burnup	*As specified in Table 2.1.25*	*As specified in Table 2.1.25*
Fuel Assembly Length	*≤ 176.8 in. (nominal design)*	*≤ 176.8 in. (nominal design)*
Fuel Assembly Width	*≤ 8.54 in. (nominal design)*	*≤ 8.54 in. (nominal design)*
Fuel Assembly Weight	*≤ 1680 lbs (including non-fuel hardware)*	*≤ 1680 lbs (including DFC and non-fuel hardware)*

Table 2.1.23 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

PARAMETER	*VALUE*
Other Limitations	▪ *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.* ▪ *Neutron sources are not authorized for storage in the MPC-24EF.* ▪ *BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.* ▪ *CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16.* ▪ *Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.*

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

PARAMETER	*VALUE (Note 1)*	
Fuel Type	*Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class*	*Uranium oxide, PWR damaged fuel assemblies and fuel debris in DFCs meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class*
Cladding Type	*ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class*	*ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class*
Maximum Initial Enrichment per Assembly	*As specified in Table 2.1.3*	*As specified in Table 2.1.3*
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU*
Decay Heat Per Fuel Storage Location	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 500 Watts*	*ZR clad: As specified in Section 2.1.9.1* *SS clad: ≤ 500 Watts*
Non-fuel hardware post-irradiation Cooling Time and Burnup	*As specified in Table 2.1.25*	*As specified in Table 2.1.25*
Fuel Assembly Length	*≤ 176.8 in. (nominal design)*	*≤ 176.8 in. (nominal design)*
Fuel Assembly Width	*≤ 8.54 in. (nominal design)*	*≤ 8.54 in. (nominal design)*
Fuel Assembly Weight	*≤ 1,680 lbs (including non-fuel hardware)*	*≤ 1,680 lbs (including DFC and non-fuel hardware)*

Table 2.1.24 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32F

PARAMETER	*VALUE*
Other Limitations	▪ *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.* ▪ *Neutron sources are not permitted for storage in MPC-32.* ▪ *BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.* ▪ *CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20.* ▪ *Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.*

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)

Post-irradiation Cooling Time (yrs)	*Inserts (Note 4) Maximum Burnup (MWD/MTU)*	*Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)*	*Control Component (Note 6) Maximum Burnup (MWD/MTU)*	*APSR Maximum Burnup (MWD/MTU)*
≥ 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)

MPC Model	*Decay Heat per Fuel Assembly (kW)*
MPC-24/24E/24EF	≤ 1.583
MPC-32/32F	≤ 1.1875
MPC-68/68FF	≤ 0.522

Table 2.1.27

MPC FUEL STORAGE REGIONS AND MAXIMUM DECAY HEAT

MPC Model	*Number of Fuel Storage Locations in Region 1* ($N_{Region\ 1}$)	*Number of Fuel Storage Locations in Region 2* ($N_{Region\ 2}$)	*Maximum Decay Heat per MPC, Q (kW)*
MPC-24/24E/24EF	*4*	*20*	*38*
MPC-32/32F	*12*	*20*	*38*
MPC-68/68FF	*32*	*36*	*35.5*

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

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

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[†] [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^3, Kr, and Xe) released in accordance with NUREG-1536.

[†] 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^3, 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^3, 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*~~a short-term~~ event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects during or immediately after, a ~~short-term~~*transient* event.

2.2.2.4 Leakage of One Seal

The *MPC enclosure vessel has 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.AN~~ 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 ~~O~~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-ST*ORM*~~AR~~ 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.

Table 2.2.3

DESIGN TEMPERATURES

HI-STORM 100 Component	*Long Term*, Normal Condition Design Temp*erature*, *Limits* (° F)	Short Term Operations, Off-Normal, and Accident Condition Temp*erature*, Limits† (° F)
MPC shell	~~450~~*500*	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
Fuel Cladding	*752*	*752 or 1058 (Short Term Operations)*†† *1058 (Off-normal and Accident Conditions)*
~~Zircaloy fuel cladding (5-year cooled)~~‡	~~691(PWR)~~ ~~740(BWR)~~	~~1058~~
~~Zircaloy fuel cladding (6-year cooled)~~‡	~~676(PWR)~~ ~~712(BWR)~~	~~1058~~
~~Zircaloy fuel cladding (7-year cooled)~~‡	~~635(PWR)~~ ~~669(BWR)~~	~~1058~~

† 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).

†† *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

HI-STORM 100 Component	*Long Term*, Normal Condition Design Temp*erature*, *Limits* (° F)	Short Term Operations, Off-Normal, and Accident Condition Temp*erature*, Limits† (° F)
~~Zircaloy fuel cladding (10 year cooled)[1]~~	~~625(PWR) 658(BWR)~~	~~1058~~
~~Zircaloy fuel cladding (15 year cooled)[1]~~	~~614(PWR) 646(BWR)~~	~~1058~~
~~Zircaloy fuel cladding (5 year cooled)[2]~~	~~679 (PWR) 740 (BWR)~~	~~1058~~
~~Zircaloy fuel cladding (6 year cooled)[2]~~	~~660 (PWR) 712 (BWR)~~	~~1058~~
~~Zircaloy fuel cladding (7 year cooled)[2]~~	~~635 (PWR) 669 (BWR)~~	~~1058~~
~~Zircaloy fuel cladding (10 year cooled)[2]~~	~~621 (PWR) 658 (BWR)~~	~~1058~~
~~Zircaloy fuel cladding (15 year cooled)[2]~~	~~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 2.2.12
STRESS LIMITS FOR DIFFERENT
LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE OVERPACK AND HI-TRAC
(ELASTIC ANALYSIS PER NF-3260)

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

Definitions:

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

† Governed by Appendix F, Paragraph F-1332 of the ASME Code, Section III.

Table 2.2.13

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

NORMAL CONDITION	
LOADING	**NOTATION**
Dead Weight	D
Handling Loads	H
Design Pressure (Internal)[‡]	P_i
Design Pressure (External)[†]	P_o
Snow	S
Operating Temperature	T
OFF-NORMAL CONDITION	
Loading	Notation
Off-Normal Pressure (Internal)[‡]	P_i'
Off-Normal Pressure (External)[†]	P_o
Off-Normal Temperature	T'
Off-Normal HI-TRAC Handling	H'

† ~~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_i^*
Accident Pressure (External)	P_o^*

Table 2.2.14
APPLICABLE LOAD CASES AND COMBINATIONS FOR EACH CONDITION AND COMPONENT[†, ††]

CONDITION	LOADING CASE	MPC	OVERPACK	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[†††], H, $P_{i\,(water\ jacket)}$
	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[†††] (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*[††††]	D, T, P_o*[††††]	D, T, P_o*[††††]
	4	N/A	D, T, (E, M, F, W')[†††††]	D, T, (M, W')[†††††]

† 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.

†† N/A stands for "Not Applicable."

††† T (normal condition) for the HI-TRAC is 100°F and $P_{i(water\ jacket)}$ 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.

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

††††† (E, M, F, W') means loads are considered separately in combination with D, T. E and F not applicable to HI-TRAC.

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 ~~a~~*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.

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[†]

[2.0.3] *Deleted.*~~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.

[†] 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.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.

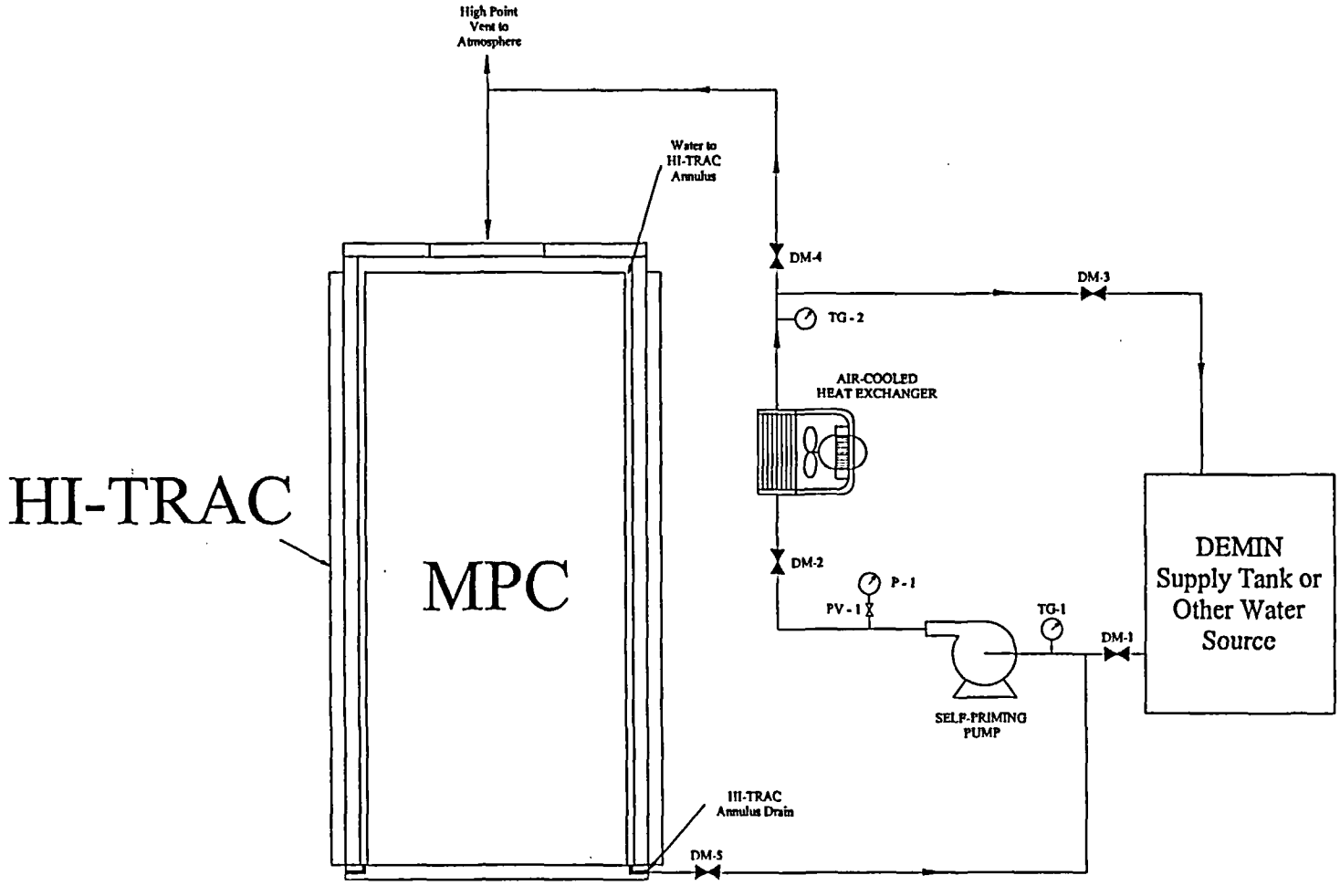

FIGURE 2.C.1: SUPPLEMENTAL COOLING SYSTEM EXAMPLE P&I DIAGRAM

CHAPTER 4† 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 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°C~~ *570°C* (~~1058°F~~ *1058°F*) for accident~~,~~ *and* off-normal *event*~~, and fuel transfer~~ conditions.

---

† 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).*

3. The maximum internal pressure of the cask should remain within its design pressures for normal (1% rod rupture), off-normal (10% rod rupture), and accident (100% rod rupture) conditions.

4. The cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.

5. 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.~~

~~7.~~ *6.* The ~~cask~~ *HI-STORM* ~~system~~ *System* should be passively cooled.

~~8.~~ *7.* 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).*

~~B~~This revision to the HI-STORM *Final* 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 ~~2~~).*

- *The Aluminum Heat Conduction Elements (AHCEs), optional under Amendment 1 of CoC 1014, are removed from the design. Removing the AHCEs from the MPC eliminates the constriction of the downcomer flow (Figure 4.0.1) and thus further enhances the thermal performance of the MPC.*

- *The whole spectrum of regionalized storage of Spent Nuclear Fuel (SNF) for each MPC type has been analyzed to allow flexibility in Region 1 (the core region of the basket) and Region 2 (the outer region of the basket) heat loads. The heat loads flexibility afforded to the ISFSI owner by the analyses documented in this FSAR permits MPCs to be loaded in the most effective manner to minimize dose.*

- *Certain elements of excessive conservatism in the mathematical model have been relaxed to retain a moderate level of conservatism. Subsection 4.4.6 documents conservatisms that apply to the thermal solution. A quantitative estimate of the consequences of the elements of conservatism is provided in Appendix 4.B.*

- *The helium backfill pressure is increased (Table 4.4.38) to facilitate increased heat dissipation from the MPC through the classical thermosiphon action (Figure 4.0.1).*

- *The design maximum decay heat load (Q_d) for the HI-STORM System is revised (Table 4.4.39).*

~~□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~~

In this chapter, the maximum HI-STORM System temperatures and pressures for normal conditions

of storage are established. The normal storage conditions are defined below:

Condition	*Value*
Decay Heat	*Values Specified in Table 4.4.39*
MPC Helium Fill Pressure	*Minimum Specified in Table 4.4.38*
Ambient Temperature	*Annual Average Temperature (Table 2.2.2)*

The HI-STORM maximum temperatures are provided in Tables 4.4.9, 4.4.10 and 4.4.26 for MPC-24/24E, MPC-68 and MPC-32 respectively and in Table 4.4.36 for the overpack temperatures. The maximum MPC pressures are provided in Table 4.4.14.

4.0.1 HI-STORM Modeling Overview

The HI-STORM 100 System is a vertical ventilated overpack having an internal cavity for emplacement of a stainless steel Multi-Purpose Canister (MPC) containing Spent Nuclear Fuel (SNF). Prior to its emplacement, the MPC internal space is pressurized with helium. The HI-STORM Overpack is equipped with four large ducts at each of its bottom and top extremities. The design of the system provides for an annular gap between the MPC and overpack. The ducted overpack construction, together with an engineered annular space between the MPC and the HI-STORM overpack enables cooling of the MPC external surfaces by ventilation action. The ventilation cooling of the HI-STORM system is illustrated in Figure 4.0.2. As shown in this figure, ambient air is drawn into the annulus from the bottom inlet ducts. The air is heated during its upward movement through the vertical annulus and exits through the top outlet ducts.

In Figure 4.0.1, a cutaway view of an MPC is shown. The fuel basket that resides in the MPC is constructed as an array of square shaped fuel cells in a honeycomb structure for storing SNF. The fuel basket (and it's stored fuel) is enclosed in an all welded pressure vessel formed by the MPC baseplate, top lid and cylindrical shell. The MPC interior space is filled with pressurized helium. The MPC design incorporates top and bottom plenums formed by elongated semi-circular holes at the base and top of fuel cell walls. Between the fuel basket and the MPC shell is an open downcomer space engineered to connect the top and bottom plenums. In this manner, the MPCs feature a fully connected helium space consisting of the fuel basket cells, top and bottom plenums and a peripheral downcomer gap.

It is heuristically apparent from the geometry of the MPC that the fuel basket metal, the fuel assemblies and the contained helium will be at their peak temperature at or near the longitudinal axis of the MPC. As a result of conduction along the metal fuel basket walls and radiant heat exchange from the fuel assemblies to the MPC fuel basket, the temperatures will attenuate with increasing radial distance from the center, reaching their lowest values in the downcomer space. As a result, the bulk temperatures of the helium columns in the fuel basket are elevated above the bulk temperature of the downcomer space. Since two fluid columns with different temperatures in communicative contact cannot remain in static equilibrium, the non-isotropic temperature field

guarantees the incipience of heat transfer by internal convection. This mode of MPC heat dissipation, "thermosiphon action," is depicted in Figure 4.0.1.

The thermosiphon action is initiated by the heating of helium residing in the fuel storage cells by the stored nuclear fuel. The heated helium expands, moving upward through the stored fuel assemblies due to buoyancy forces. The upward moving helium exits into the top plenum space turning 90° as shown in Figure 4.0.1. The helium flows through the top plenum towards the MPC shell, turns 90° and flows down in the downcomer space. In this space, the helium is progressively cooled as it flows down, rejecting its heat to the MPC shell. At the bottom of the downcomer, the helium turns 90° into the bottom plenum space to supply the fuel basket cells with cooled helium. In this manner a circulation of helium is sustained inside the MPC by the heat from the stored fuel.

From the discussion above, it is apparent that the thermal model of the HI-STORM system must include the following elements:

i) HI-STORM overpack
ii) MPC with it's enclosed fuel basket
iii) Air-flow in the HI-STORM annular space
iv) Helium circulation inside the MPC

A thermal model of the HI-STORM System incorporating the elements listed above is shown in Figure 4.0.3. In addition, as we explain next, the model recognizes the "coupled" nature of the heat transfer process in the HI-STORM System wherein the internal circulation of helium inside the MPC occurs with the upward flow of air outside of it. The MPC shell, also shown in this figure, is the interface boundary that separates the two fluids – helium and air. At this interface boundary air is moving upward on the outside of the MPC shell and helium is moving downward. Heat is exchanged from hot fluid (helium) to cold fluid (air) across a metal boundary, which is similar to that of a counter-flow heat exchanger. The heat exchange progressively lowers the helium temperature in the down-flow direction and progressively elevates that of air in the up-flow direction. It is apparent that the internal circulation of helium influences the heat input profile to the annulus air. The helium circulation removes heat from the fuel cells and rejects a portion of it to the upper elevations of the MPC shell and to the MPC lid. As a result, heat input to the air is skewed towards the upper portion of the MPC shell. Based on classical chimney operating principles, adding more heat towards the top of an air stack is less efficient for air circulation relative to adding heat at lower elevations.

In order to properly simulate the interaction of the helium circulation and the annulus air flow, a "coupled fluid" thermal model is constructed for the HI-STORM thermal analyses. The coupled fluid thermal model includes both fluids (helium and air) in one unified model. The model is constructed on the FLUENT version 6.1 computer code from FLUENT Inc., a software company based in Lebanon, NH. The modeling methodology is described in Subsections 4.4.1.1.2 through 4.4.1.1.9. The results of the analysis are presented in Subsection 4.4.2 and evaluated in Subsection 4.4.6.

4.1 DISCUSSION

As discussed in Chapter 2, this revision of the HI-STORM FSAR seeks to establish complete compliance with the provisions of Reference [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 now 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 CSF. 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 Reference [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~~ *in Figure 4.0.2.* ~~(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~~ *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 *4.4.3*~~81.2.2~~). 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 *4.3.1* ~~4.3.7~~) 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~~ *Section 4.0.1 describes* ~~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 multi-cellular 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~~ *Three* distinct MPC basket geometries are evaluated for thermal performance in the HI-

STORM System. For intact PWR fuel storage, the *24-cell flux trap (*MPC-24*/24E)*, ~~MPC-24E,~~ and *32-cell non-flux trap (*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 *three* ~~four~~ basic MPC geometries (MPC-32, MPC-24*/24E* ~~, 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~~ *listed in Table 4.4.39 for single region storage (also referred to as uniform storage)*. Storage of intact stainless steel 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 surfaces 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 *in Subsection 4.4.6 and Appendix 4.B,* ~~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 *more* 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_2, 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.V.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 benchmarking 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 t*T*hermal m*M*odel and its b*B*enchmarking with f*F*ull*Size* size c*C*ask t*T*est d*D*ata" *[4.1.5]*, Holtec Report HI-992252, Rev. 1, documents the comparison of the Holtec thermal model against the full-size test data [4.1.3]~~. In *this benchmarking report,* ~~rReference [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*s* a

comparison of the Holtec thermal model results with the PNL solution.. ~~Once again,~~ *The comparison shows that* the 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.

4.3 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 demonstrates that* ~~performance that demonstrates that~~ fuel cladding temperatures are maintained below design basis limits. *Neutron absorber materials* ~~Boral~~ used in MPC baskets for criticality control (~~a composite material composed of~~ *made from* B_4C and aluminum) ~~is~~ *are* stable up to 1000°F[†] ~~for short-term and 850°F for long-term dry storage~~[‡]. 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 thatAmerican 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* ~~under~~ *long term normal, short term operations,* off-normal ~~or~~ *and* ~~hypothetical~~ accident conditions ~~, material temperature limits for short-duration events~~ are provided in Table 4.3.1. *In Table 4.3.1, ISG-11, Rev. 3 temperature limits [4.1.4] are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials that are approved by the NRC for power generation.*

4.3.1 Subsection Intentionally Deleted

4.3.2 Subsection Intentionally Deleted

4.3.3 Subsection Intentionally Deleted

4.3.4 Evaluation of MBF

It is recognized that hydrides present in irradiated fuel rods (pre-dominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.8]. Upon cooling below a threshold temperature (T_p), the hydrides precipitate and re-orient to an undesirable (radial) direction if cladding stresses at the hydrides precipitation temperature T_p are excessive. For moderate burnup

---

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

~~‡ AAR Advanced Structures Boral thermophysical test data.~~

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 re-orientation under hydride precipitation temperatures for MBF [4.3.8]. The study concludes that hydride re-orientation is not credible during short term operations involving low to moderate burnup fuel (upto 45 GWD/MTU). Accordingly, the higher ISG-11, Rev. 3 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).

Table 4.3.1

HI-STORM SYSTEM MATERIAL TEMPERATURE LIMITS

Material	**Normal Long-Term Temperature Limits [°F]**	**Short Term *Operations, Off-Normal and Accident* Temperature Limits [°F]**
~~Zircaloy fuel cladding~~ *CSF Cladding (Zirconium alloys and Stainless Steel)*	~~(Moderate‡ Burnup)~~ ~~See Table 4.3.7~~ *752*	*Short Term Operations:* *752°F (HBF)* *1058°F (MBF)* *Off-Normal and Accident:* *1058°F*
~~Stainless steel fuel cladding~~	~~806~~	~~1058~~
~~Boral~~§- *Neutron Absorber*	800	950
Holtite-A†††	~~300~~ *N/A*	~~300~~ *350 (Short Term Operations)*
Concrete**	~~200~~ *300*	350
Water	~~307~~ *N/A*	*307†† (Short Term Operations)* N/A *(Off-Normal and Accident)*

~~‡ High burnup fuel storage limits are established in Appendix 4.A.~~

~~§ Based on AAR Structures Boral thermophysical test data.~~

††† See Section 1.2.1.3.2.

** *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.*

†† *Saturation temperature at HI-TRAC water jacket design pressure.*

Tables 4.3.2 through 4.3.9 are 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. The details of the thermal models, analyses and results for long term storage scenarios are presented in this section. An overview of the methodology for characterizing the thermal-hydraulic properties of the MPC and a description of the global HI-STORM model used for obtaining the HI-STORM temperature field are provided. The modeling details are discussed in Subsections 4.4.1.1.2 through 4.4.1.1.9. As discussed in Subsection 4.0.1, the HI-STORM model incorporates the "coupled-fluid" approach needed to solve the thermo-fluid interaction of two fluids in the HI-STORM system, viz. air in the annulus and helium in the MPC.

4.4.1 Thermal Model

The MPC basket design consists of three 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 neutron absorber sandwiched between the box wall and a stainless steel sheathing plate over the full length of the active fuel region.

To ensure a robust margin in the HI-STORM system, the MPC property characterizations employ a composite of limiting thermal-hydraulic characteristics among each of two fuel classes (PWR and BWR fuel). These characteristics are: (a) An upperbound planar thermal resistance, (b) a lowerbound axial thermal conductivity and (c) an upperbound axial flow resistance. The fuel types that obtain the limiting result for each of the three characteristics (a), (b) and (c) above are listed below for ready reference.

I) Fuel Class: PWR Fuel
   (a) Planar thermal resistance: W 17x17 OFA
   (b) Axial thermal conductivity: W 14x14 OFA
   (c) Axial flow resistance: B&W 15x15

II) Fuel Class: BWR Fuel
   (a) Planar thermal resistance: GE-11 9x9
   (b) Axial thermal conductivity: GE 7x7
   (c) Axial flow resistance: GE-12/14 10x10

The design basis decay heat generation (Q_d) is defined in Table 4.4.39. 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).

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 using three 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 & Epstein [4.4.1] formulation (which considered radiative heat exchange between fuel rod surfaces).

2. The second model considers heat transport within an MPC fuel basket 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. To model MPC internal convection heat transfer, the fuel basket is rendered as a porous media having effective flow resistance characteristics.

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 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, namely:

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.

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 an MPC fuel basket 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) . There are four distinct materials in this section, namely the homogenized fuel cell , the Alloy X structural materials (including neutron absorber sheathing), neutron absorber, and helium gas. Each of the 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 fuel basket 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 fuel basket is influenced by radiation, which is a temperature-

dependent effect, the equivalent conductivity of the MPC fuel basket 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 fuel 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. 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, 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. The thermal modeling duly recognizes the MPC as a non-isotropic 3-D conduction media. In particular, the MPC fuel basket in-plane conductivity and the axial conductivity differ considerably. This is because while in-plane heat transfer is interrupted by gaps, in the axial direction heat flows in an uninterrupted manner in the fuel rods, the fuel cell walls and neutron absorber plates. Accordingly, the thermal properties of the MPC fuel basket continuum are characterized by two conductivities:

(i) Planar conductivity (K_r) as a function of temperature
(ii) Axial conductivity (K_{ax}) as a function of temperature

The procedure for computing K_r and K_{ax} is described in Subsection 4.4.1.1.4. As stated earlier in this section, the MPC thermal property evaluations conservatively employ limiting fuel types for computing K_r and K_{ax}. These limiting fuel types are listed earlier in Subsection 4.4.1. Finally, to model internal convection in the MPC, the fuel storage cells are rendered as porous media having effective pressure drop characteristics. The methodology for modeling flow resistance is discussed next.

For simulating the flow resistance of fuel cells, the FLUENT porous media pressure drop model is employed. In this model, pressure drop is computed (in MKS units) as follows:

$$\frac{\Delta P}{L} = \frac{\mu}{\alpha} V + C\left(\frac{1}{2}\rho V^2\right)$$

where:

$\Delta P/L$ is pressure drop per unit length (Pa/m)
V is superficial fluid velocity (m/s)
μ is fluid viscosity (kg/m-s)
ρ is fluid density (kg/m^3)
α is permeability parameter (m^2)

C is inertial resistance factor (m^{-1})

The PWR and BWR fuel assemblies occupying the fuel cells in an MPC are essentially an array of regularly spaced fuel rods supported by an open grid structure at discrete locations. From basic hydraulics, pressure drop in a fuel cell is obtained as the sum of two parts: (i) frictional loss from the array of fuel rods and (ii) pressure losses from fluid contractions and expansions at the grid locations. For maximizing frictional losses, the thermal analysis employs the limiting fuel type for which the smallest hydraulic diameter is obtained. For maximizing grid spacer flow resistances, two assumptions are employed:

a) Spacers are a gridwork of thick (0.05 inch) plates
b) Bounding expansion and contraction loss factors are used

The first assumption above understates flow area at the grid spacer location and the second assumption overstates hydraulic losses. Together the assumptions ensure a substantial conservatism in the fuel assembly pressure drop characteristics.

Employing basic principles of fluid mechanics, the frictional and grid hydraulic losses are computed and appropriate values of the parameters (α and C) in the porous media pressure drop model obtained. The flow resistances are computed for the limiting fuel type‡‡ in each class (PWR or BWR fuel). The limiting fuel types were listed earlier in Section 4.1.1. The principal steps for computing "α" and "C" are provided next.

Step 1: Compute Hydraulic Diameter (D_h)

$$D_h = \frac{4A_f}{WP}$$

where:

A_f is cell flow area (cell opening area minus area of fuel rods)
WP is wetted perimeter (sum of rods circumference and cell walls perimeter)

Step 2: Compute Porosity (ε)

$$\varepsilon = \frac{A_f}{A_o}$$

where A_o is cell area defined by the square of cell pitch

Step 3: Compute "α"

$$\alpha = \frac{\varepsilon D_h^2}{32}$$

‡‡ The limiting fuel type has the smallest hydraulic diameter.

Step 4: Compute Cumulative Grids Loss Factor (K)

$$K = N\,(Kc + Ke)$$

where:

N is number of flow grids
Kc is flow contraction coefficient
Ke is expansion coefficient

Step 5: Compute "C"

$$C = \frac{K}{\varepsilon_g^2 L}$$

where:

L is fuel cell length
ε_g is ratio of grids flow area to cell area A_o

The porous media pressure model described in the foregoing is benchmarked with proprietary pressure drop data to confirm that the model predictions are conservative. As stated previously, the flow resistance parameters (α and C) are obtained for the limiting fuel type in each class (PWR or BWR fuel) for use in the HI-STORM thermal evaluations.

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.
In this manner, a loaded MPC standing upright on the ISFSI pad in a HI-STORM overpack is replaced in the model as 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 In-Plane 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 through a screening process using the simplified procedure described below. This is followed by the determination of temperature-dependent properties for 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 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 first establish which assembly type has the maximum in-plane thermal resistance. The same determination is made for the MPC-68, out of the 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. To identify the CSF with the highest in-plane thermal resistance the model proposed by Wooton and Epstein [4.4.1] is used.

According to [4.4.1], transport of heat energy within any cross section of a fuel assembly occurs through 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:

$$Q = \sigma C_o F_\varepsilon A [T_C^4 - T_B^4] + 13.5740 L K_{es} [T_C - T_B]$$

where:

F_ε = Emissivity Factor

$$= \frac{1}{\left(\frac{1}{\varepsilon_C} + \frac{1}{\varepsilon_B} - 1\right)}$$

ε_C, ε_B = emissivities of fuel cladding, fuel basket (see Table 4.2.4)

C_o = Assembly Geometry Factor

$$= \frac{4N}{(N+1)^2} \text{ (when N is odd)}$$

$$= \frac{4}{N+2} \text{ (when N is even)}$$

N = Number of rows or columns of rods arranged in a square array
A = fuel assembly "box" heat transfer area = 4 × width × length
L = fuel assembly length
K_{cs} = fuel assembly constituent materials volume fraction weighted mixture conductivity
T_C = hottest fuel cladding temperature (°R)
T_B = box temperature (°R)
Q = net radial heat transport from the assembly interior
σ = Stefan-Boltzmann Constant (0.1714×10^{-8} Btu/ft^2-hr-°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 conductivities of the fuel types for several representative PWR and BWR assemblies are 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). Based on this simplified analysis, the W-17×17 OFA PWR and GE11-9×9 BWR fuel assemblies are determined to be the bounding fuel types for planar conductivity of zircaloy clad fuel.

Having established the governing (most resistive) PWR and BWR SNF types, we use a finite-volume code to determine the effective conductivities in a rigorous 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. As discussed later in this subsection, the models depicted in these figures have been benchmarked with results from independent technical sources to confirm that they yield conservative results.

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 (ΔT) from the center of the cross section to the periphery is analytically given by [4.4.5]:

$$\Delta T = 0.29468 \frac{q_g L^2}{K}$$

This analytical formula is applied to determine the effective material conductivity from a known quantity of heat generation applied in the FLUENT model (smeared as a uniform heat source, q_g) basket opening size and ΔT calculated in the first step.

As discussed earlier, the effective fuel space conductivity 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.

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 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 neutron absorbing plate for criticality control. Each neutron absorber plate is sandwiched in a sheathing-to-basket wall pocket. A schematic of the "Box Wall- Neutron absorber-Sheathing" sandwich geometry of an MPC basket is illustrated in Figures 4.4.6 and 4.4.7. During fabrication, a uniform normal pressure is applied to each "Box Wall-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 gaps. The mean coefficient of linear expansion of the 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 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 neutron absorber and box-sheathing surfaces. A conservative contact resistance resulting from a 2 mil neutron absorber to pocket gap is applied in the analysis. In other words, no credit is taken for the interfacial pressure between neutron absorber and stainless plate/sheet stock produced by the fixturing and welding process.

Heat conduction properties of a composite "Box Wall- 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, - helium gap, neutron absorber 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-Sheathing" sandwich in ANSYS. A fixed temperature (T_c) 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 (T_h) of the fixed-flux edge. The equivalent thermal conductivity is the obtained using the resulting temperature difference across the sandwich as input to a one-dimensional Fourier equation as follows:

$$K_{eff} = \frac{q \times L}{T_h - T_c}$$

where:

K_{eff} = effective thermal conductivity
q = heat flux applied in the ANSYS model
L = ANSYS model heat transfer path length
T_h = ANSYS calculated average edge temperature
T_c = specified edge temperature

The heat transfer path length will vary, depending on the direction of transfer (i.e., in-plane or out-of-plane).

4.4.1.1.4 Modeling of Fuel 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 reported herein .

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- 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" will 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 equivalent thermal properties in the in-plane and axial directions is discussed in this subsection.

(i) In-Plane Conductivity of the Fuel Basket

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. 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 as stated earlier in Subsection 4.4.1 ( ) 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. 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 evaluation in which the sixteen outer cells are occupied with damaged fuel, the effect of reduced conductivity on the PCT is negligible (~ 1°F). Therefore, DFCs do not pose a limitation on safe storage of fuel. The fuel basket planar conductivities are provided in Table 4.4.3.

(ii) Axial Conductivity of the Fuel Basket

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). In Table 4.4.3 the MPC fuel basket planar and axial conductivities are provided.

4.4.1.1.5 Subsection Intentionally Deleted

4.4.1.1.6 Subsection Intentionally Deleted

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. For this purpose, a "coupled-fluid" model of the HI-STORM System is constructed for all HI-STORM analyses. The model, depicted in Figure 4.0.3 and discussed in Subsection 4.0.1 includes both fluids, viz. air (in the annulus) and helium (in the MPC), in one model. The model incorporates the interaction between the annulus air and the hot helium circulating inside the MPC.

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 ($F_{o\text{-}i}$) is given by [4.4.3]:

$$F_{o\text{-}i} = \frac{1}{R} - \frac{1}{\pi R} \times [\cos^{-1}(\frac{B}{A}) - \frac{1}{2L}\{\sqrt{(A+2)^2 - (2R)^2} \times \cos^{-1}(\frac{B}{RA}) + B\sin^{-1}(\frac{1}{R}) - \frac{\pi A}{2}\}]$$

where:

$F_{o\text{-}i}$ = 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 ($F_{i\text{-}a}$) from outer overpack surface, represented by the inner cylinder, to the ambient can be determined as:

$$F_{i\text{-}a} = 1 - F_{o\text{-}i}\frac{R_o}{R_i}$$

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 porous 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 described in Subsection 4.4.1.1.2 provide 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 fuel types.

Having replaced the fuel spaces by porous 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, 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 region whose thermal conductivity is a strong function of temperature.

The thermal model for the HI-STORM overpack is prepared as a three-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 – a classic design (HI-STORM 100) and an enhanced version (HI-STORM 100S) overpack The HI-STORM 100 and HI-STORM 100S overpacks are thermally similar in as much as they yield thermal solutions that are in reasonable agreement (~5°F). .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 (ε) 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 by assuming a bounding absorbtivity of unity. .

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. A summary of the essential features of this model is presented in the following:

- A lower bound canister pressure of 105 psia under normal operating condition is postulated.
- 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 a horizontal plane for uniform loading and in each planar region (inner and outer) in regionalized storage. and varies in the axial direction in accordance with 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.

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.

As discussed in Chapter 2 of this FSAR, the commercial nuclear fuel (CSF) can be stored in any of the MPCs in a uniform or regionalized configuration. The uniform storage arrangement implies that the heat emission rate of every fuel assembly (intact or canisterized) must be less than or equal to the MPC heat duty Q_d (Table 4.4.39) divided by the number of storage locations. In regionalized storage, the storage locations in the fuel basket are divided into two regions, denoted as Region 1 and Region 2, respectively. The cells located in the core region of the basket constitute Region 1. The peripheral cells define Region 2. The regionalization of the fuel basket must be configured such that a fuel assembly located in Region 1 is completely surrounded by Region 2 cells. Thus the Region 2 fuel serves as an effective blocker of gamma radiation emitted by the fuel located in Region 1. Regionalizing the storage of SNF such that the low heat emitting (therefore, lower dose emitting fuel) is located in Region 2 and the high heat emitting fuel is confined to the core of the basket (Region 1) provides an effective method to minimize the radiation emitted from canister at a storage pad or in transport. In particular, the dose limits prescribed for HLW packages in 10CFR71 mandate that careful attention be paid to minimize dose emitted from an MPC. By regionalizing fuel storage in each basket, the aggregate dose at the site boundary can also be minimized which has a direct relevance to public health and safety. Figures 4.4.27, 4.4.28 and 4.4.29 depict Region 1 and 2 arrangements for MPC-32, MPC-24/24E, and MPC-68 respectively. In Table 4.4.30, the cell number

allocations for regionalized storage are provided. In order to minimize the dose from the MPC the design basis heat emission rate of fuel in Region 2 must be as low as practical. Denoting N_1 and N_2 respectively as the number of storage cells in regions 1 and 2, it follows that the MPC heat load (Q) in regionalized storage is:

$$Q = N_1 q_1 + N_2 q_2$$

where q_1 and q_2 are the per assembly heat generation rates in Regions 1 and 2 respectively. For reasons that we explain below, when $q_1 = q_2$ (uniform storage), the value of Q is the maximum, say Q_d. In fact Q decreases monotonically as the ratio $X = q_1/q_2$ increases. Stated differently, the greater the disparity in the specific heat generation rates in Regions 1 and 2, the greater the penalty on the heat duty of the MPC. The reason that $Q < Q_d$ for regionalized storage§§ becomes clear by considering the temperature profile in a typical MPC. As can be seen from Figures 4.4.19 and 4.4.20, the maximum cladding temperature is at its peak value at the centerline (core) of the basket and decreases in the manner of a parabola with the radial co-ordinate. In other words, the design basis heat duty of the basket, Q_d, is governed by the permitted value of the Peak Cladding Temperature (PCT) which occurs at the centerline of the MPC. If the uniform storage (equal heat generation rate in each cell) is replaced by two regions and the MPC decay heat heat non-uniformly distributed in a manner that the core region is populated with more heat emissive fuel and outer region with relatively less emissive fuel, then clearly the center line temperature will be further elevated. To meet the PCT limit, the heat generation rate in Region 2 must reduced resulting in an overall reduction in the heat duty of the MPC. To be sure, increasing the rate of heat generation in the core region of the MPC has a salutary effect of increasing the thermosiphon driven upflow of helium through the core region. However, calculations show that the enhanced heat transfer through a more vigorous helium circulation is not enough to cancel out the elevation in PCT due to locally increased heat generation.

To summarize, 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. 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 the outer region with fuel of heat load q_2, where $q_1 > q_2$.. As depicted in Figures 4.4.27, 4.4.28 and 4.4.29 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 storage scenario.

As discussed previously, under regionalized storage the MPC heat load (Q), defined previously, is monotonically reduced as the disparity between the inner and outer region heat loads defined by the parameter X (equal to q_1/q_2) is increased. Following this logic path, the design heat loads under regionalized storage are defined by a mathematical function Q(X) that satisfies two attributes:

§§ The reduction in Q due to regionalization is the direct outcome of placing comparatively hotter fuel in the core region. The opposite arrangement, wherein the hotter fuel is located in Region 2, would permit the heat duty of the MPC to be increased beyond Q_d. Placing more heat emissive fuel in the outer region, however would be anti-ALARA and therefore precluded as a storage option in the HI-STORM system.

1) Q(X) reduces to uniform storage design heat load (Q_d) as X approaches unity
2) Q(X) monotonically reduces with increasing X

A function Q(X) satisfying (1) and (2) above is adopted for regionalized storage. The mathematical expression for Q(X) is:

$$Q(X) = \frac{2Q_d}{(1+X^a)}$$

where a is a constant (= 0.15) and X is permitted to vary from a minimum value of 1 to a maximum value of 6. Graphical plots of Q(X) are provided in Figures 4.4.31 and 4.4.32 for PWR and BWR MPCs, respectively. To confirm that the ISG-11, Rev. ~~3~~2 cladding temperature limits are met for the design heat loads stated in the manner above, an array of HI-STORM thermal analyses are performed over a broad range of X (from 1 to 6) for each MPC type. From these analyses the variation in peak cladding temperature with X is obtained, results reported and evaluated in Subsection 4.4.2.

For the physical problem of regionalized storage, an infinite number of combinations of q_1 and q_2 exist that would satisfy the PCT limits of the stored fuel. To provide maximum flexibility to the ISFSI owner, an array of thermal analyses for each MPC type are performed to establish a continuum of regionalized storage options defined by the decay heats ratio X and the design heat load function Q(X). The permissible regionalized heat loads are computed by a four step process:

Step 1: Select a numerical value of X between 1 and 6.

Step 2: Compute maximum permissible MPC heat load:
$Q = 2Q_d/(1+X^a)$

Step 3: Compute permissible Region 2 SNF heat generation:
$q_2 = Q/(N_1X + N_2)$

Step 4: Compute permissible Region 1 SNF heat generation:
$q_1 = q_2X$

4.4.1.1.10 Subsection Intentionally Deleted

4.4.1.1.11 Subsection Intentionally Deleted

4.4.1.1.12 Subsection Intentionally Deleted

4.4.1.1.13 Subsection Intentionally Deleted

4.4.1.1.14 MPC Helium Backfill Pressure

The quantity of helium emplaced in the MPC cavity shall be sufficient to produce an operating pressure of 105 psia during normal storage at the design basis heat load. Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain minimum helium backfill pressure (P_b) specified at a reference temperature (70°F). The minimum backfill pressure for each MPC type is provided in Table 4.4.37. A theoretical upper limit on the helium backfill pressure also exists and is defined by the design pressure of the MPC vessel (Table 2.2.1 in Chapter 2). The upper limit of P_b is reported in the last column of Table 4.4.37. To bound the minimum and maximum backfill pressures listed in Table 4.4.37 with a margin, a helium backfill specification for all MPCs is set forth in Table 4.4.38.

There are two methods available for ensuring that the appropriate quantity of helium has been placed in an MPC:

i. By pressure measurement
ii. By measuring the quantity of MPC helium backfill (in standard cubic feet)

The direct pressure measurement approach is more convenient if the FHD method of MPC drying is used. In this case, a certain quantity of helium is already in the MPC. Because the helium is mixed inside the MPC during the FHD operation, the temperature of the helium gas at the MPC's exit, along with the pressure provides a reliable means to compute the inventory of helium in the MPC cavity using pressure and temperature gages. The pressure in the FHD system, after adjustment from the measured helium temperature to the reference temperature (70°F), is adjusted by addition or withdrawal of helium such that it lies in the mid-range of the P_b specifications.

When vacuum drying is used as the method for MPC drying, then it is more convenient to fill the MPC by introducing a known quantity of helium (in standard cubic feet) by measuring the quantity of helium introduced using a calibrated mass flow meter. The required quantity of helium (F) is computed by the product of net free nominal volume and helium specific volume at a target pressure that lies in the mid-range of the P_b specifications.

The net free nominal volume of the MPC is obtained by subtracting A from B, where

A = MPC cavity volume in the absence of contents (fuel and non-fuel hardware) computed from nominal design dimensions

B = Total volume of the contents (fuel including DFCs, if used) based on nominal design dimensions

Using commercially available mass flow totalizers, an MPC cavity is filled with the computed quantity of helium (F).

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 conservatisms 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 principal 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. Both uniform loading and regionalized loading scenarios are analyzed. For uniform loading, the MPC thermal payload is assumed to be at the design maximum (Q_d defined in Section 4.0). For regionalized loading, the storage scenarios are defined by the ratio $X = q_1/q_2$ and an associated maximum heat load Q(X) defined in Subsection 4.4.1.1.9. The HI-STORM System is analyzed under regionalized storage for X ranging from 1 to 6.

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 in-plane 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 in-plane 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, 4.4.10 and 4.4.26 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, representative axial temperature plots of the fuel cladding are shown in Figures 4.4.16 and 4.4.17 for a MPC-24 and a 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 and 4.4.26 which also summarize maximum calculated temperatures in different parts of the MPCs and HI-STORM overpack (Table 4.4.36).

In Figures 4.4.19 and 4.4.20, respectively, representative radial temperature distributions in an MPC-24 and an MPC-68 in the horizontal plane where the maximum fuel cladding temperature occurs are graphed. Finally, axial variations of the ventilation air temperatures and that of the overpack inner shell surface are graphed in Figure 4.4.26 for a bounding MPC. .

As stated in Subsection 4.4.1.1.9 an array of HI-STORM analyses are performed for the regionalized storage option to characterize the variation of peak cladding temperature (PCT) with the decay heats ratio X. The PCT variations are graphed in Figures 4.4.33 through 4.4.35 over the permitted range of X. From these graphs it is evident that the PCT is below the ISG-11, Rev. *3* ~~2~~-limits for all MPCs.

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, Rev. *3* ~~2~~) under all storage scenarios (uniform and regionalized) and all MPCs.
- The maximum temperature of the basket structural material is within the stipulated design temperature.
- The maximum temperature of the neutron absorber is below the design temperature 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 NRC's recommended limits [4.4.10] (See Table 4.3.1.)

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 with a follow up discussion in Appendix 4.B. 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 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 (BPRA 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_4C in a refractory solid matrix (Al_2O_3). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin-walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operation conditions (2,300 psia and approximately 600°F coolant temperature). The B_4C- Al_2O_3 neutron absorber material is principally used in B&W and CE fuel BPRA designs. The relatively low temperature of the poison material in BPRA rods (relative to fuel pellets) favors 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 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 sub-section, 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 and 4.4.26).

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 heat up 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 4.4.30 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 and 4.4.26 for the maximum computed basket periphery temperatures). To maximize the fuel basket differential expansion, the basket periphery-to-MPC shell temperature difference is conservatively maximized ($\Delta T = 175°F$). From the bounding temperature profile T(r) and ΔT, the mean fuel basket temperature (T1) and MPC shell temperature (T2) are computed as follows:

$$T1 = \frac{\int_0^1 rT(r)dr}{\int_0^1 rdr}$$

$$= 676°F$$

$$T2 = TP - \Delta T$$
$$= 425°F$$

The differential radial growth of the fuel basket (Y1) from an initial reference temperature (To = 70°F) is computed as:

$$Y1 = R*\{A1*(T1 - To) - A2*(T2 - To)\}$$

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 4.4.40 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. A comprehensive discussion of HI-STORM conservatisms is provided in Appendix 4.B. A numbered list of the many thermal modeling conservatisms for long-term storage is provided hereunder:

1. The most severe levels of environmental factors for long-term normal storage, which are an ambient temperature of 80°F and 10CFR71 insolation levels, were coincidentally imposed on the system.

2. The most adversely located*** HI-STORM System in an ISFSI array was considered for analysis.

3. The thermosiphon effect in the MPC, which is intrinsic to the HI-STORM fuel basket design is included in the thermal analyses, using a conservative model, by assuming the top and bottom plenum spaces for helium flow to be less than actual values.

4. 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.

5. The MPC is assumed to be loaded with an SNF which has the maximum resistance to heat dissipation (planar and axial) and helium flow of all fuel types in its category (BWR or PWR), as applicable.

6. 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.

7. Conservative bounding flow resistance factors are employed to simulate flow through MPC 3-D continuum.

8. Axial heat transfer through fuel pellets is ignored.

9. Heat dissipation by grid spacers, top & bottom fittings is ignored.

10. Insolation heating assumed with a bounding absorbtivity (=1.0).

11. A margin between the computed peak cladding temperature and 400°C limit is provided for all MPCs.

12. The computed values of MPC axial conductivities are understated in the thermal models.

13. The flux trap flow areas (MPC-24/24E designs) are ignored in the MPC thermosiphon cooling models.

*** 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 environment. To bound these effects, a reference cask is enclosed in a hypothetical reflecting cylinder as described in Section 4.4.1.1.9 and Appendix 4B.

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 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

Fuel	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
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† - 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

† Haddam Neck Plant B&W or Westinghouse stainless steel clad fuel assemblies.

Table 4.4.2

SUMMARY OF BWR FUEL ASSEMBLY EFFECTIVE THERMAL CONDUCTIVITIES

Fuel	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
Dresden 1 - 8×8†	0.119	0.201	0.319
Dresden 1 - 6×6†	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††-10×10 SS	0.152	0.222	0.309
Exxon-10×10 SS	0.151	0.221	0.308
Damaged Dresden-1 8×8† (in a Holtec damaged fuel container)	0.107	0.169	0.254
Humboldt Bay-7x7†	0.127	0.215	0.343
Dresden-1 Thin Clad 6x6†	0.124	0.212	0.343
Damaged Dresden-1 8x8 (in TN D-1 canister)†	0.107	0.168	0.252
8x8 Quad+ Westinghouse†	0.164	0.276	0.435

† 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.

†† Allis-Chalmers stainless steel clad fuel assemblies.

Table 4.4.3

MPC BASKET PLANAR AND AXIAL[†††] THERMAL CONDUCTIVITY VALUES

Basket	@200°F (Btu/ft-hr-°F)	@450°F (Btu/ft-hr-°F)	@700°F (Btu/ft-hr-°F)
MPC-24/24E	1.127 (planar) 2.427 (axial)	1.535 (planar) 2.666 (axial)	2.026 (planar) 2.870 (axial)
MPC-68	1.025 (planar) 2.186 (axial)	1.257 (planar) 2.379 (axial)	1.500 (planar) 2.548 (axial)
MPC-32	0.964 (planar) 1.773 (axial)	1.214 (planar) 1.933 (axial)	1.486 (planar) 2.079 (axial)

††† The axial thermal conductivity values reported herein are understated 5%.

Table 4.4.4 through 4.4.8

[INTENTIONALLY DELETED]

Table 4.4.9

HI-STORM† SYSTEM LONG-TERM NORMAL
UNIFORM STORAGE MAXIMUM TEMPERATURES
(MPC-24/24E‡‡‡ BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	688	752§§§
MPC Basket****	633	725†††
Basket Periphery	569	725†††
MPC Shell	466	500

† Bounding overpack temperatures are provided in Table 4.4.36.

‡‡‡ Limiting values reported.
§§§ The temperature limit is in accordance with ISG-11, Rev. 3 ~~2~~.
**** The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.
††† 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† SYSTEM LONG-TERM NORMAL
UNIFORM STORAGE MAXIMUM TEMPERATURES
(MPC-68 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	731	752††
MPC Basket††††	706	725†††
Basket Periphery	579	725†††
MPC Shell	470	500

† Bounding overpack temperatures are provided in Table 4.4.36.

†† The temperature limit is in accordance with ISG-11, Rev. *3* ~~2~~.

†††† The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.

††† 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

INTENTIONALLY DELETED

Table 4.4.12

SUMMARY OF MPC-24 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	45
Bounding Fuel Assemblies Volume	79
Basket Supports and Fuel Spacers Volume	7
Net Free Volume	236 (6683 liters)

Table 4.4.13

SUMMARY OF MPC-68 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	35
Bounding Fuel Assemblies Volume	93
Basket Supports and Fuel Spacers Volume	12
Net Free Volume	227 (6428 liters)

Table 4.4.14
SUMMARY OF MPC CONFINEMENT BOUNDARY PRESSURES†
FOR LONG-TERM STORAGE

Condition	Pressure (psig)‡‡‡‡
MPC-24:	
Initial backfill (at 70°F)	48.8
Normal condition	97.9
With 1% rods rupture	98.6
With 10% rods rupture	104.8
With 100% rods rupture	167.1
MPC-68:	
Initial backfill (at 70°F)	48.8
Normal condition	97.3
With 1% rods rupture	97.7
With 10% rods rupture	101.7
With 100% rods rupture	141.5
MPC-32:	
Initial backfill (at 70°F)	48.8
Normal Condition	97.0
With 1% rods rupture	98.0
With 10% rods rupture	106.6
With 100% rods rupture	192.3
MPC-24E:	
Initial backfill (at 70°F)	48.8
Normal Condition	97.9
With 1% rods rupture	98.6
With 10% rods rupture	105.0
With 100% rods rupture	169.3

† Per NUREG-1536, pressure analyses with postulated rods rupture (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.

‡‡‡‡ The pressures reported in this table are computed assuming the helium backfill pressure is at it's upperbound limit (Table 4.4.38).

Table 4.4.15 through 4.4.21

[INTENTIONALLY DELETED]

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, $I_{1/2}$	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:

N_T - Maximum Annual Average (Normal) Temperature (80°F)
O_T - Off-Normal Temperature (100°F)
E_T - Extreme Hot Temperature (125°F)
Q_D - Design Basis Maximum Heat Load
SS - Steady State
SS(B) - Bounding Steady State
TA - Transient Analysis
AH - Adiabatic Heating

I_O - All Inlet Ducts Open
$I_{1/2}$ - Half of Inlet Ducts Open
I_C - All Inlet Ducts Closed
ST - Insolation Heating (Top)
SC - Insolation Heating (Curved)
F - Fire Heating (1475°F)

Table 4.4.23

[INTENTIONALLY DELETED]

Table 4.4.24

SUMMARY OF MPC-32 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume.	25
Bounding Free Assemblies Volume	106
Basket Supports and Fuel Spacers Volume	9
Net Free Volume	227 (6428 liters)

Table 4.4.25

SUMMARYOF MPC-24E FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	52
Bounding Fuel Assemblies Volume	79
Basket Supports and Fuel Spacers Volume	7
Net Free Volume	229 (6484 liters)

Table 4.4.26

HI-STORM† SYSTEM LONG-TERM NORMAL UNIFORM STORAGE
MAXIMUM TEMPERATURES
(MPC-32 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	662	752††
MPC Basket§§§§	621	725†††
Basket Periphery	565	725†††
MPC Shell	463	500

† Bounding overpack temperatures are provided in Table 4.4.36.

†† The temperature limit is in accordance with ISG-11, Rev. *3* ~~2~~.

§§§§ The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.

††† 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 through 4.4.29

[INTENTIONALLY DELETED]

Table 4.4.30

MPC REGIONALIZED STORAGE CONFIGURATIONS

MPC Type	Inner Region Assemblies (n_1)	Outer Region Assemblies (n_2)	Depicted in Figure
MPC-24/24E	4	20	Figure 4.4.28
MPC-32	12	20	Figure 4.4.27
MPC-68	32	36	Figure 4.4.29

Table 4.4.31 through 4.4.35

[INTENTIONALLY DELETED]

Table 4.4.36

BOUNDING LONG-TERM NORMAL STORAGE
HI-STORM OVERPACK TEMPERATURES

Component†	Local Section Temperature†† (°F)	Long-Term Temperature Limit (°F)
Inner shell	243	350
Outer shell	183	350
Lid bottom plate	412	450
Lid top plate	218	450
MPC pedestal plate	180	350
Baseplate	108	350
Overpack Body Concrete ~~Radial shield~~	193	300
Overpack Lid Concrete	*295*	*300*
Overpack Pedestal Concrete	*142*	*300*

† See Figure 1.2.8 for a description of HI-STORM components.
†† Section temperature is defined as the through-thickness average temperature.

Table 4.4.37

ACCEPTABLE RANGE OF MPC HELIUM BACKFILL PRESSURE*****

MPC	Minimum Backfill Pressure [psig]	Maximum Backfill Pressure [psig]
MPC-32	45.0	50.5
MPC-24/24E	44.5	50.0
MPC-68	44.9	50.4

***** The pressures tabulated herein are at a reference gas temperature of 70°F.

Table 4.4.38

MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS

Item	Specification
Minimum Pressure	45.2 psig @ 70°F Reference Temperature
Maximum Pressure	48.8 psig @ 70°F Reference Temperature

Table 4.4.39

DESIGN MAXIMUM HEAT LOADS FOR THE HI-STORM SYSTEM

MPC Type	Heat Load (Q_d) [kW]
PWR MPCs	38
BWR MPCs	35.5

Table 4.4.40

SUMMARY OF HI-STORM DIFFERENTIAL EXPANSIONS

Gap Description	Cold Gap (U), inch	Differential Expansion (V), inch	Is Free Expansion Criteria Satisfied (i.e. U > V)
Fuel Basket-to-MPC Radial Gap	0.1875	0.096	Yes
Fuel Basket-to-MPC Axial Gap	1.25	0.499	Yes
MPC-to-Overpack Radial Gap	0.5	0.139	Yes
MPC-to-Overpack Axial Gap	1.0	0.771	Yes

4.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS‡

4.5.1 Synopsis of Short Term Operations

Prior to placement in a HI-STORM overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transferred 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. As stated in Chapter 2, ISG-11, Rev. *3* ~~2~~ places a temperature limit on the fuel cladding temperature under all short term operations.

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 design platforms denoted as HI-TRAC 100 and HI-TRAC 125. 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 of the two HI-TRAC designs, the permissible temperature limits specified in Reference [4.1.4] ~~,~~ ~~unless otherwise justified,~~ 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. Loading Operations with Flooded MPC
ii. Drying of the MPC Cavity
iii. Onsite transport in HI-TRAC
iv. MPC Cooldown and Reflood for Defueling Operations

Each of the above conditions corresponds to a distinct thermal state for the spent fuel stored in the MPC. The first three conditions pertain to MPC loading operations; the last scenario is germane to the rare case when a loaded MPC needs to be defueled. Out of the four scenarios listed above, demoisturization using the vacuum drying method is thermally most challenging causing the greatest elevation in fuel cladding temperatures. On-site 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 wherein on-site transport of an MPC must occur with the HI-TRAC in the horizontal configuration. Horizontal transfer is thermally more adverse than its vertical configuration because of the suppression of the thermosiphon mode of cooling when HI-TRAC is horizontal. The above mentioned scenarios associated with fuel loading are carried out in reverse if the contents of an MPC have to be unloaded.

The fuel handling operation scenarios described above place a certain level of constraint to the dissipation of heat from the MPC relative to the normal storage condition. Consequently, for some scenarios, it is necessary to limit the maximum MPC heat generation rates to those at

‡ Because of the significant quantity of new material required to satisfy ISG-11 *Revision 3* ~~(latest revision)~~, this section has been rewritten in its entirety.

which the steady state fuel cladding temperature limits are approached. Henceforth, these limiting heat generation rates for each scenario will be referred to as the "threshold heat load" (Q_T) for that scenario. For certain scenarios the threshold heat loads may be more restrictive than the design basis heat load. The threshold heat loads to comply with the prescribed temperature limits for the short term operations are computed and reported in this chapter. Analyses are performed for short term operations, and threshold heat loads obtained and reported in Subsections 4.5.4 through 4.5.7.

4.5.1.1 Supplemental Cooling System

For a HI-TRAC loaded with greater than threshold heat loads, 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 including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the supplemental cooling system will be satisfied by any plant-specific design are set forth in FSAR Chapter 2 as Appendix 2.C.

4.5.2 Thermal Acceptance Criteria for Short Term Operations

As stated in Section 4.1, the HI-STORM FSAR seeks to establish complete compliance with the provisions of Reference [4.1.4]. This requires the maximum cladding temperature to not exceed ISG-11 limits (see Table 4.3.1) during short term operations for all high burnup fuel.

~~As stated in Section 4.1, for MPCs loaded with only Moderate Burnup Fuel (MBF), short term operations are permitted provided the following two criteria are satisfied[§]:~~

~~(i)The estimated cladding hoop stress, σ_{max}, does not exceed 90 MPa.~~
~~(ii)The peak cladding temperature is below 570°C (1058°F).~~

~~Because the cladding stress is a function of cladding thickness and cladding corrosion, the estimation of stress (Criterion (ii) above), of necessity, must be fuel-specific. The necessary computations are performed by a cask user opting for this criterion by employing the methodology described next.~~

~~The procedure for estimating σ_{max} to establish compliance with Criterion (ii) above is carried out in the following steps:~~

~~a. Determine the axial temperature distribution in the hottest rod using the HI-STORM thermal model presented in this chapter.~~

~~The applicable heat load is based on the SNF batch to be loaded in the MPC. The maximum SNF heat generation rate (calculated using methods described in Chapter 5) is~~

~~§ For MBF that does not muster compliance with the σ_{max} criteria, the Reference [4.1.4] temperature limit shall apply.~~

~~ascribed to every fuel assembly to bound the actual cumulative heat generation rate from all SNF in an MPC.~~

~~b. Compute the average gas temperature in the hottest rod~~

~~For computing the average gas temperature in the fuel rod, two distinct axial zones in the fuel rod are identified. One zone is the fuel rod pellets stack region wherein the gas space consists of the annular gap between the pellet and the cladding inside surface. The other region is the gas plenum space above the fuel pellets. T_{avg} is obtained by a gas volume weighted average of the mean rod temperature in the two regions. T_{avg} is computed for the hottest fuel rod by the following formula:~~

$$T_{avg} = \frac{v\frac{1}{a}\int_0^a T(z)dz + \frac{V}{L-a}\int_a^L T(z)dz}{v+V}$$

~~where: T(z): axial rod temperature profile~~
~~v: gas volume in pellet-to-clad gap~~
~~V: plenum gas volume~~
~~a: length of pellet stack region~~
~~L: fuel rod length~~

~~c. Compute the plenum gas pressure, P_v~~

~~The initial thermodynamic state of gas confined inside a fuel rod is specified by two parameters: Gas pressure (P_o) and a reference gas temperature (T_o) in absolute units. By the Ideal Gas Law, the rod gas pressure upon heat up under vacuum (P_v) is proportional to the average gas temperature of a fuel rod (T_{avg}). In other words, $P_v = P_o (T_{avg}/T_o)$.~~

~~d. Compute the maximum cladding hoop stress, σ_{max}~~

~~The hoop stress (σ_{max}) developed in cladding is a function of rod internal diameter (d_i) cladding thickness (t) and an internal rod gas pressure (P_v). The stress is computed by the Lame formula given below:~~

$$\sigma_{max} = \frac{P_v d_i}{2t}$$

~~The cladding thickness should be taken as the nominal thickness less the estimated metal loss due to corrosion. The inside diameter of the cladding should be taken as the nominal diameter. Satisfying the above cited 90 MPa limit is an essential requirement for the MBF's cladding temperature limit to be set at 570°C for short term operations.~~

4.5.3 The HI-TRAC Thermal Model

4.5.3.1 Overview

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 site . Section views of the HI-TRAC are provided 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 modes of heat transfer.

Two HI-TRAC transfer cask versions, namely HI-TRAC 100 and HI-TRAC 125 have the same design features but with lower member thicknesses in the 100 ton version to yield a lighter weight design. The analytical model developed for HI-TRAC thermal characterization is constructed to bound both 100 and 125 ton HI-TRAC transfer casks. In this manner, the HI-TRAC overpack resistance to heat transfer is overestimated, resulting in higher predicted MPC internals and fuel cladding temperature levels.

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. A small diametral 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. Each water jacket cavity is long and narrow having the cross section of an annular sector. Conduction heat transfer occurs through both the water cavities and the radial connectors . 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 cask is in contact with a supporting surface. This face is conservatively modeled as an insulated surface. Because 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. Summary descriptions of the various components of HI-TRAC's thermal model are provided below.

4.5.3.2 Effective Thermal Conductivity of Water Jacket

The HI-TRAC water jacket is composed of an array of radial ribs equispaced around the HI-TRAC body and affixed to enclosure plates by welding to form discrete water compartments. Holes in the radial ribs 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 a minimum available metal thickness for radial heat transfer. 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

$N_r t_r$ = Available metal thickness (product of number of radial ribs and rib thickness)
L_R = effective radial heat transport length through water spaces
K_w = thermal conductivity of water
$N_r t_w$ = Cumulative waterspaces width (between radial ribs)

4.5.3.3 Heat Rejection from Transfer Cask Exterior Surfaces

The following relationship for the surface heat flux from the outer surface of an isolated cask to the environment applied to the thermal model:

$$q_s = 0.19 (T_s - T_A)^{4/3} + 0.1714\varepsilon [(\frac{T_s + 460}{100})^4 - (\frac{T_A + 460}{100})^4]$$

where:

T_S = cask surface temperatures (°F)
T_A = ambient atmospheric temperature (°F)
q_s = surface heat flux (Btu/ft^2×hr)
ε = 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×Pr) numbers exceeds 10^9. This product can be expressed as $L^3 \times \Delta T \times Z$, where L is the characteristic length, ΔT is the surface-to-ambient temperature difference, and Z is a function of the surface temperature (defined in Section 4.2). The characteristic length of a vertically oriented HI-TRAC is its height of approximately 17 feet. The value of Z, conservatively taken at a bounding surface temperature of 340°F, is 2.6×10^5. Solving for the value of ΔT that satisfies the equivalence $L^3 \times \Delta T \times Z = 10^9$ yields $\Delta T = 0.78$°F. For a horizontally oriented HI-TRAC the characteristic length is the diameter of approximately 7.6 feet (minimum of 100- and 125-ton versions), yielding $\Delta T = 8.76$°F. The natural convection will be turbulent, therefore, provided the surface to air temperature difference is greater than or equal to 0.78°F for a vertical orientation and 8.76°F for a horizontal orientation.

4.5.3.4 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. Twelve-hour averaged insolation levels are prescribed in 10CFR71. 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 24-hour averaged insolation levels.

4.5.3.5 Lead-to-Steel Interface

Lead, poured between the inner and outer shells of the HI-TRAC body, is utilized as a gamma shield material in the HI-TRAC transfer cask. Unlike many metal cask designs that utilize pre-fabricated lead "bricks", lead is installed in the HI-TRAC transfer cask in molten form. The lead pouring process is a mature technology and proven methods to preclude internal voids or gaps are well established in the industry.

To ensure a homogeneous lead pour, the HI-TRAC shell is pre-heated and molten lead is poured to fill the annular cavity. Ladling of the molten lead aided by the pressure from the column of molten lead helps ensure that internal voids or separation would not occur. The lead installation includes appropriate inspection measures, including gamma scan and weighing of the cask after lead placement, to confirm that the lead column installed in the HI-TRAC cask are without internal voids or gaps. Therefore, the lead-to-steel interface is assumed to be an uninterrupted continuum in the HI-TRAC thermal model.

4.5.4 Loading Operations with Flooded MPC

The HI-TRAC containing an MPC loaded with fuel and filled with water is removed from the cask pit (which may or may not be integral to the fuel pool) and placed in a designated space

(typically on the pool deck) that is henceforth referred to as the Decontamination and Assembly Station (DAS). The early operations that occur at the DAS include welding of the main lid and testing of the welded joint using methods described in Chapter 8.

To minimize personnel dose and to keep the SNF in a cooled state, the SNF is kept submersed in water until the fuel drying operation (discussed in the next subsection) is initiated. The temperature of the fuel cladding is not a concern in the operational evolutions with the flooded MPC. In accordance with NUREG-1536, however, boiling of water in the MPC cavity is not permitted during wet loading operations. This requirement is met 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 draining and fuel drying operations.

When the HI-TRAC transfer cask containing the loaded water-filled MPC is removed from the pool, or when the MPC lid is installed in 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 made:

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 is credited in the analysis. The 100-ton version has a significantly smaller quantity of metal mass, which will result in a higher rate of temperature rise.

iv. A conservatively bounding MPC cavity free volume is considered for flooded water mass.

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)

C_h = combined thermal inertia of the loaded HI-TRAC transfer cask (Btu/°F) [see Table 4.5.5]
T = temperature of the contents (°F)
t = time after HI-TRAC transfer cask is removed from the pool (hr)

A heat-up rate for the HI-TRAC transfer cask contents is determined employing a bounding design heat load (Table 4.4.39, Q = 1.3×10^5 Btu/hr) is determined as 4.99 °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:

$$t_{max} = \frac{T_{boil} - T_{initial}}{(dT/dt)}$$

where:

T_{boil} = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)
$T_{initial}$ = 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 temperatures.

In a situation where the maximum allowable time provided in Table 4.5.6 is insufficient to complete all wet fuel handling operations, a suitable means of heat removal such as a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water is introduced via the MPC lid drain port connection and heated water exits 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

As an illustrative example, if the MPC cavity water temperature is limited to 150°F, assuming an MPC inlet water temperature of 125°F and design maximum heat load, the water flow rate is computed as 5200 lb/hr (10.4 gpm). The required minimum flow rate shall be calculated using the actual MPC heat load and the temperature of the cooling water available for this operation.

4.5.5 MPC Drying

4.5.5.1 Drying Options

This FSAR provides for two methods for drying Commercial Spent Fuel (CSF) the MPC cavity, namely:

i. Forced Helium Dehydration
ii. Vacuum Drying

The methods are discussed next.

4.5.5.2 Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a closed loop dehumidification system consisting of a condenser, a demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. Appendix 2.B contains 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 in Table 4.3.1 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 storage. As a result, the peak fuel cladding temperatures will approximate the values reached during normal storage as described elsewhere in this chapter.

4.5.5.3 Vacuum Drying

Because the vacuum drying method of demoisturization leads to a considerable rise in the fuel cladding temperature, threshold heat load limits** that are considerably lower than the MPC design basis heat load are computed for the vacuum drying evolution. The threshold heat loads are very low if one or more high burnup fuel (HBF) assemblies are included in the batch of fuel being loaded. If the fuel batch consists of only MBF~~, and the limitations on the maximum cladding stress described in Subsection 4.5.2 are met~~, then a higher threshold heat load meeting

** See threshold heat load discussion in Subsection 4.5.1

the less restrictive temperature limit is permitted.

(a) Analysis

The vacuum condition effective fuel assembly conductivity is determined by procedures discussed earlier (Subsection 4.4.1.1.2) with recognition of the attenuation of thermosiphon effect with the decrease in the quantity of helium and reduction in the conductivity of helium in a most conservative manner. For this purpose, the thermal conductivity of fluid media is set at a miniscule fraction of the helium conductivity. A direct result of this assumption is that the cladding temperatures are exaggerated in the thermal solutions and, correspondingly, threshold heat loads understated. . 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. The heat transported to the MPC shell is dissipated from the external surface of the MPC shell to the annulus. It is recognized that the cladding temperature is directly affected by the temperature of the annulus. To ensure a robust margin in the cladding temperatures, vacuum drying is performed with a water cooled annulus. Two options are provided for annulus water cooling:

(i) Standing water in the annulus.
(ii) Annulus gap is water flushed.

An axisymmetric FLUENT thermal model of the MPC in a HI-TRAC is constructed, and peak cladding temperatures at threshold heat loads are obtained. The following conditions are applied to this evaluation:

i. The fuel temperatures rise to their asymptotic steady-state values.

ii. The outer surface of the MPC shell is postulated to be at a bounding temperature of 232°F†† (standing water in annulus) or 125°F‡‡ (continuously flushed with water).

iii. The bottom surface of the MPC is insulated.

(b) Results

Table 4.5.11 provides the threshold heat loads at or below which for which vacuum drying is permitted. For completeness, the threshold heat load under the FHD method of drying is also listed (it is equal to the design heat load). The threshold heat load under the vacuum drying condition is a function of two parameters:

†† Saturation temperature of water at the bottom of a water filled HI-TRAC annulus.
‡‡ During vacuum operations, water must be circulated at a rate sufficient to ensure maximum annulus temperature is below 125°F. For example given water inlet at 100°F, Q=15 kW and a flush rate of 5 gpm, the maximum temperature (from an adiabatic heat balance for Q = 15 kW) is 120.5°F which is below 125°F.

i. Maximum burnup in the fuel batch stored
ii. The MPC-HI-TRAC annulus is stagnant or water flushed

As ~~stated earlier,~~ the permissible temperature for a fuel batch containing only MBF ~~can be as high as~~ *is higher* ~~570°C~~ *(See Table 4.3.1), the threshold heat loads are also correspondingly greater* ~~if the clad stress criterion in Subsection 4.5.2 is met~~. The maximum fuel cladding temperature is quite obviously influenced by the thermal state in the annulus: continuous flushing helps reduce the peak cladding temperature. Table 4.5.11 accordingly provides discrete values of the threshold heat loads depending on annulus condition (standing water or annulus flushing) and fuel burnup.

4.5.6 On-site Transport in HI-TRAC

4.5.6.1 Analysis

An axisymmetric FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was constructed to evaluate temperature distributions for onsite transport scenarios for two HI-TRAC annulus conditions:

(a) Static column of air
(b) Water cooled annulus

Steady-state analyses of the HI-TRAC transfer cask have been performed for the two annulus conditions under all on-site transport scenarios and threshold heat loads obtained. 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.

4.5.6.2 Results

As stated earlier, the threshold heat loads depend on the orientation of the HI-TRAC. The threshold heat load for vertical transport are greater than that for horizontal transport. Another variable that affects the computed threshold heat load for the on-site transport condition is the maximum burnup in the batch of fuel loaded in the MPC. Finally, if the actual heat generation rate in the MPC exceeds the threshold heat load permitted for the HI-TRAC orientation and burnup state of the CSF batch loaded then annulus cooling is required. Table 4.5.12 provides the threshold heat loads for all on-site transport scenarios.

For both horizontal and vertical mode of on-site transfer of the loaded MPC in the HI-TRAC transfer cask, threshold heat generation rates to meet the HBF fuel clad temperature limit are well below the design basis heat load for the MPC under the normal condition of storage. These threshold heat loads are provided in Table 4.5.12 for vertical and horizontal mode of on-site MPC transfer under steady state conditions. If the heat load of a canister exceeds the threshold value listed in Table 4.5.12, then supplemental cooling of the MPC must be provided to maintain the peak fuel cladding temperature below limit set forth in this FSAR.

Because of the narrow annular space between the MPC and HI-TRAC transfer cask and the availability of a threaded coupling connection in the bottom HI-TRAC lid, it is possible to provide augmented heat removal from the MPC by circulating a coolant through the annular space during MPC transfer operations. Calculations show that even when the threshold heat loads are substantially exceeded, a modest flow of water[§§] is all that is needed to extract sufficient amount of heat to ensure that the peak cladding temperature is below the ISG-11 Rev. *3* ~~2~~ limit adopted in this FSAR. For example, the principal variables and results from an evaluation performed for an MPC-32 at its design basis heat load are summarized in Table 4.5.13 to illustrate the concept. As shown in this table, the peak cladding temperature in the cooled MPC is much below the ISG-11 Rev. *3* ~~2~~ limit.

Because the availability of utilities (demineralized water, compressed air, etc.) is plant-specific, it is not possible to design a standard ancillary that can be used at all sites. Nevertheless, the above example serves to demonstrate that the equipment required to effect the necessary heat removal to subcool the MPC during transfer operation for high heat load MPCs will be quite compact and operationally expedient.

It is seen from the above example that even a modest means to cool the external surface of the MPC during on-site transport is sufficient to create a substantial margin in the peak cladding temperature against the permissible limit. This margin may be necessary in certain plants to deal with a short-term handling step during the transfer operation when it may not be practical to maintain auxiliary cooling. Such a situation may occur at certain plants, for example, when the HI-TRAC transfer cask is being positioned over the HI-STORM overpack for the MPC's transfer. In the absence of the auxiliary cooling, if the canister heat load exceeds the Table 4.5.12 threshold heat load, then the fuel cladding temperature will begin to rise. To ensure that the amount of cladding temperature rise is not enough to cause an exceedance of the permissible temperature limit, the MPC must be sufficiently pre-cooled prior to the start of the transfer step when the external cooling becomes unavailable. Calculations show that, with appropriate pre-cooling, a reasonable amount of time to execute an operational step (with the external cooling turned off) can be provided.

To illustrate the MPC heat-up scenario, the water cooled MPC summarized in Table 4.5.13 is analyzed with the HI-TRAC rotated to the horizontal configuration (thermally most adverse configuration) and the auxiliary cooling system turned off. At the beginning of the thermal transient, the MPC is assumed to be in the thermal state condition analyzed above corresponding to the steady state condition with the HI-TRAC vertical and the external cooling operative. The rise in the peak cladding temperature as a function of time is shown in Figure 4.5.4 for this case. This figure shows that 7-1/2 hours are required for the peak cladding temperature to reach 752°F. If a longer period of time were warranted by the operational step, the auxiliary cooling system would be sized to ensure that, prior to initiation of the operation with auxiliary cooling withdrawn, the fuel cladding will be cooled below the applicable clad temperature limit by the required amount. The extent of required pre-cooling, of course, will depend on the MPC heat

§§ For scenarios that exceed the threshold heat loads by modest amounts, a static column of water in the annulus is adequate.

load, orientation of the HI-TRAC, and the expected duration of the operational step. However, as the above example illustrates, the approach to pre-cool the MPC to maintain the peak cladding temperature below the regulatory limit during a short-time, cooling-unaided operational step is quite feasible.

HI-TRAC cask temperature results are reported for thelimiting sceanario that obtains the highest cladding temperature (Condition 3 and 7, Table 4.5.12). The results are provided in Table 4.5.2 summarizing the maximum calculated temperatures in different parts of the HI-TRAC transfer cask and MPC.

4.5.7 MPC Cooldown and Reflooding for Defueling Operations

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. For high heat load MPCs, the extremely rapid cooldown rates to which the hot MPC internals and the fuel cladding can be subjected during water injection may, however, result in high thermal stresses. Additionally,, water injection may result in large quantities of steam generation. To protect the fuel cladding from high thermal strains under direct water quenching the MPCs are cooled using appropriate means prior to the introduction of water in the MPC cavity space.

Because of the continuous gravity driven circulation of helium in the MPC which results in heated helium gas in sweeping contact with the underside of the top lid and the inner cylindrical surface of the enclosure vessel, utilizing an external cooling means to remove heat from the MPC is quite effective. The external cooling process can be completely non-intrusive such as extracting heat from the outer surface of the enclosure vessel using chilled water. Extraction of heat from the external surfaces of an MPC is very effective largely because of the thermosiphon induced internal transport of heat to the peripheral regions of the MPC. The non-intrusive means of heat removal is preferable to an intrusive process wherein helium is extracted and cooled using a closed loop system such as a Forced Helium Dehydrator (Appendix 2.B), because it eliminates the potential for any radioactive crud to exit the MPC during the cooldown process. Because the optimal method for MPC cooldown is heavily dependent on the location and availability of utilities at a particular nuclear plant, mandating a specific cooldown method cannot be prescribed in this FSAR. Simplified calculations are presented in the following to illustrate the feasibility and efficacy of utilizing an intrusive system such as a recirculating helium cooldown system.

Under a closed-loop forced helium circulation condition, the helium gas is cooled, via an external chiller. The chilled helium is then introduced into the MPC cavity from connections at the top of the MPC lid. The helium gas enters the MPC basket and moves through the fuel basket cells, removing heat from the fuel assemblies and MPC internals. The heated helium gas exits the MPC from the lid connection to the helium recirculation and cooling system. Because of the turbulation and mixing of the helium contents in the MPC cavity by the forced circulation, the MPC exiting temperature is a reliable measure of the thermal condition inside the MPC cavity.

The objective of the cooldown system is to lower the bulk helium temperature in the MPC cavity to below the normal boiling temperature of water (212°F). For this purpose, the rate of helium circulation shall be sufficient to ensure that the helium exit gas temperature is below this threshold limit with a margin.

An example calulation for the required helium circulation rate is provided below to limit the helium temperature to 200°F. The calculation assumes no heat loss from the MPC boundaries and a design maximum heat load*** (1.3×10^5 Btu/hr). Under these assumptions, the MPC helium is heated adiabatically by the MPC decay heat from a given inlet temperature (T1) to a temperature (T2). The required circulation rate to limit T2 to 200°F is computed as follows:

$$m = \frac{Q_d}{C_p(T2 - T1)}$$

where:

Q_d = Design maximum decay heat load (Btu/hr)
m = Minimum helium circulation rate (lb/hr)
Cp = Heat capacity of helium (1.24 Btu/lb-°F (Table 4.2.5))
T1 = Helium supply temperature (assumed 15°F in this example)

Substituting the values for the parameters in the equation above, m is computed as 567 lb/hr.

4.5.8 Minimum Temperatures for On-Site Transport

In Table 2.2.2, 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, all HI-TRAC materials will satisfactorily perform their intended functions at this minimum postulated temperature condition.

4.5.9 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:

*** Table 4.4.39 lists MPC design heat loads. From this Table, the maximum heat load (38 kW) is used in this evaluation.

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 absorbtivity 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 (Table 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. Expected margins during normal HI-TRAC use will be larger due to the many conservative assumptions incorporated in the analysis. Corresponding MPC internal pressure evaluation shows that the MPC confinement boundary remains well below the short-term condition design pressure. 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 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 to retain pressure up to 60 psig 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.

Table 4.5.1

[INTENTIONALLY DELETED]

Table 4.5.2

HI-TRAC TRANSFER CASK STEADY-STATE
MAXIMUM TEMPERATURES FOR LIMITING SCENARIOS

Component	High Burnup Fuel (Condition 3, Table 4.5.12) Temperature [°F]
Fuel Cladding	745
MPC Basket	728
Basket Periphery	620
MPC Outer Shell Surface	433
HI-TRAC Overpack Inner Surface	305
Water Jacket Inner Surface	286
Enclosure Shell Outer Surface	265
Water Jacket Bulk Water	246
Axial Neutron Shield†	288

† Local neutron shield section temperature.

Table 4.5.3 and 4.5.4

[INTENTIONALLY DELETED]

Table 4.5.5

SUMMARY OF LOADED 100-TON HI-TRAC TRANSFER CASK
BOUNDING COMPONENT
WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
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†	6,500	1.0	6,500
			26,032 (Total)

† Conservative lower bound water mass.

Table 4.5.6

MAXIMUM ALLOWABLE TIME DURATION FOR WET TRANSFER OPERATIONS

Initial Temperature††† (°F)	Time Duration (hr)
115	19.4
120	18.4
125	17.4
130	16.4
135	15.4
140	14.4
145	13.4
150	12.4

††† Pool water temperature during fuel loading.

Tables 4.5.7 through 4.5.10

INTENTIONALLY DELETED

Table 4.5.11

THRESHOLD HEAT LOADS FOR FUEL DRYING

Condition No.	Drying Method	Threshold Heat Load (Q_T)	Burnup State	Is Annulus Flush Required?	Cladding Temperature Limit [°F)	Computed Maximum Cladding Temperature [°F]
1	FHD	Q_d‡‡‡	MBF	No	1058	Note 1
2	FHD	Q_d	HBF	No	752	Note 1
3	VD	9 kW	HBF	No	752	729
4	VD	10 kW	HBF	Yes	752	740
5	VD	17 kW	MBF	No	1058	1042
6	VD	18 kW	MBF	Yes	1058	1056

Note 1: Under the FHD method for MPC drying, an externally driven circulation of helium ensures drying conditions in the MPC, which are in the neighborhood of the saturation temperature of water at the prevailing pressure (about 350°F). As such the operating clad temperatures are substantially below the cladding temperature limit (752°F) thereby ensuring a hospitable thermal environment for HBF.

Acronyms:
FHD – Forced Helium Dehydration
VD – Vacuum Drying
MBF – Moderate Burnup Fuel
HBF – High Burnup Fuel

‡‡‡ Design heat load (Table 4.4.39).

Table 4.5.12

PERMISSIBLE HEAT LOADS FOR ON-SITE TRANSPORT

Condition No.	HI-TRAC Orientation	Threshold Heat Load (Q_T)	Fuel burnup	Is Annulus Cooling Required?	Temperature Limit (°F)	Computed Maximum Cladding Temperature [°F]
1	Horizontal	19 kW	HBF	No	752	735
2	Horizontal	Q_d[§§§]	HBF	Yes	752	Note 1
3	Vertical	23 kW	HBF	No	752	745
4	Vertical	Q_d	HBF	Yes	752	Note 1
5	Horizontal	30 kW	MBF	No	1058	983
6	Horizontal	Q_d	MBF	Yes	1058	Note 1
7	Vertical	Q_d	MBF	No	1058	1017

Note 1: The maximum cladding temperature when annulus cooling is required will be dependent on site specifics such as decay heat of loaded SNF and availability of cooling fluid (air or water). Each user shall confirm that the annulus cooling provided is sufficient to keep the cladding temperatures below their applicable limits. An annulus cooling example is provided in Subsection 4.5.6.2.

Acronyms:
FHD – Forced Helium Dehydration
VD – Vacuum Drying
MBF – Moderate Burnup Fuel
HBF – High Burnup Fuel

§§§ Design heat load (Table 4.4.39).

Table 4.5.13

HI-TRAC ANNULUS COOLING EXAMPLE

MPC Type	MPC-32
Design Heat Load	38 kW
Transfer Cask Orientation	Vertical
Threshold Heat Load per Table 4.5.12	23kW
Coolant (Water) Flow Rate (assumed)	10 gpm
Coolant Inlet Temperature, °F (assumed)	80
Coolant Outlet Temperature, °F (calculated)	106
Peak Fuel Cladding Temperature, °F (calculated)	385

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. ~~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.~~ *For long term normal conditions ~~and off-normal operations,~~ the maximum CSF cladding temperature is below the ISG-11 [4.1.4] limit of 400°C (752°F).* ~~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 ~~Normal Handling and Onsite Transfer~~ *Short Term Operations*

As discussed in Section 4.1, 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) 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 NUREG-1536 (4.0,IV,2),~~ *As required by ISG-11* the fuel cladding temperature is maintained below *the applicable temperature limis for HBF and MBF fuel (Table 4.3.1) during* ~~400°C (752°F) 570°C (1058°F) for fuel transfer~~ *short term* operations. Maximum clad temperatures ~~for *short term operations*~~ *(on-site transport and vacuum drying conditions)* ~~normal on-site transfer conditions~~ are reported in Sections *4.5.6 and 4.5.5 respectively.* ~~4.5.2. Maximum~~ *The* clad temperatures ~~for vacuum drying conditions are reported in Section 4.5.2.1 and~~ *for short term operations* comply *with the ISG-11 limits.* ~~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 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 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.~~

4.7 REFERENCES

[4.1.1] ANSYS Finite Element Modeling Package, Swanson Analysis Systems, Inc., Houston, PA, 1993.

[4.1.2] FLUENT Computational Fluid Dynamics Software, Fluent, Inc., Centerra Resource Park, 10 Cavendish Court, Lebanon, NH 03766.

[4.1.3] "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses," EPRI NP-5128, (April 1987).

[4.1.4] "Cladding Considerations for the Transportation and Storage of Spent Fuel", Interim Staff Guidance – 11, Revision 3.

[4.1.5] "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.

[4.2.1] Baumeister, T., Avallone, E.A. and Baumeister III, T., "Marks' Standard Handbook for Mechanical Engineers," 8th Edition, McGraw Hill Book Company, (1978).

[4.2.2] Rohsenow, W.M. and Hartnett, J.P., "Handbook of Heat Transfer," McGraw Hill Book Company, New York, (1973).

[4.2.3] Creer et al., "The TN-24P Spent Fuel Storage Cask: Testing and Analyses," EPRI NP-5128, PNL-6054, UC-85, (April 1987).

[4.2.4] Rust, J.H., "Nuclear Power Plant Engineering," Haralson Publishing Company, (1979).

[4.2.5] Kern, D.Q., "Process Heat Transfer," McGraw Hill Kogakusha, (1950).

[4.2.6] "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, (August 1981).

[4.2.7] "Safety Analysis Report for the NAC Storable Transport Cask," Docket No. 71-9235.

[4.2.8] ASME Boiler and Pressure Vessel Code, Section II, Part D, (1995).

[4.2.9] Jakob, M. and Hawkins, G.A., "Elements of Heat Transfer," John Wiley & Sons, New York, (1957).

[4.2.10] ASME Steam Tables, 3rd Edition (1977).

[4.3.1] *Deleted* ~~Levy, I.S., et al., "Recommended Temperature Limits for Dry Storage of Spent~~

~~Light Water Reactor Zircaloy Clad Fuel Rods in Inert Gas," PNL-6189, (May 1987).~~

[4.3.2] *Deleted* ~~Johnson, Jr., A.B. and Gilbert, E.R., "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases," PNL-4835, (September 1983).~~

[4.3.3] *Deleted* ~~"Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," Regulatory Guide 3.54, Revision 1, (January 1999).~~

[4.3.4] *Deleted* ~~Cunningham et. al., "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," EPRI TR-106440, (April 1996).~~

[4.3.5] *Deleted* ~~Schwartz, M.W., Witte, M.C., Lawrence Livermore National Laboratory, "Spent Fuel Cladding Integrity During Dry Storage," UCID-21181.~~

[4.3.6] *Deleted* ~~"Temperature Limit Determination for the Inert Dry Storage of Spent Nuclear Fuel," EPRI TR-103949, (May 1994).~~

[4.3.7] *Deleted* ~~Schemel, J.H., "ASTM Manual on Zirconium and Hafnium," STP 639, American Society for Testing and Materials, (December 1977).~~

[4.3.8] Lanning, D.D. and Beyer C.F, "Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rods During Short Term Operations for Dry Cask Storage", PNNL white paper, (January 2004).

[4.4.1] Wooton, R.O. and Epstein, H.M., "Heat Transfer from a Parallel Rod Fuel Element in a Shipping Container," Battelle Memorial Institute, (1963).

[4.4.2] Rapp, D., "Solar Energy," Prentice-Hall, Inc., Englewood Cliffs, NJ, (1981).

[4.4.3] Siegel, R. and Howell, J.R., "Thermal Radiation Heat Transfer," 2nd Edition, McGraw Hill (1981).

[4.4.4] *Deleted* ~~Holman, J.P., "Heat Transfer," 6th ed., McGraw Hill Book Company, (1986).~~

[4.4.5] Sanders et al., "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements," Sandia Report SAND90-2406, TTC-1019, UC-820, page II-127, (November 1992).

[4.4.6] *Deleted* ~~Hewitt, G.F., Shires, G.L. and Bott, T.R., "Process Heat Transfer," CRC Press, (1994).~~

[4.4.7] *Deleted* ~~Hagrman, Reymann and Mason, "MATPRO-Version 11 (Revision 2) A~~

~~Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, Tree 1280, Rev. 2, EG&G Idaho, August 1981.~~

[4.4.8] *Deleted* ~~"Effective Thermal Conductivity and Edge Conductance Model for a Spent Fuel Assembly," R. D. Manteufel & N. E. Todreas, Nuclear Technology, 105, 421–440, (March 1994).~~

[4.4.9] *Deleted* ~~"Spent Nuclear Fuel Effective Thermal Conductivity Report," US DOE Report BBA000000-01717-5705-00010 REV 0, (July 11, 1996).~~

[4.4.10] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," USNRC, (January 1997).

[4.4.11] *Deleted* ~~"Fuel Cladding Cladding Temperatures in Transport and Storage Casks Development and Validation of a Computation Method," S. Anton, Ph.D. Thesis (German) RWTH Aachen, Germany, 1997.~~

[4.4.12] *Deleted* ~~"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.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~~ *narrow* 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 *4.0.1.* ~~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~~ *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 turbulation 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 *thermal design* ~~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 <u>average</u> ambient temperature for the <u>entire</u> 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

* 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~~ *embedded* 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 *as* ~~for a 20-year period. Based~~ *based* on meteorological data, the 80°F ~~is chosen to~~ bound*s* 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

* 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') *is* 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~~ *MODELING THE MPC TOP PLENUM*~~BASKET AXIAL RESISTANCE~~

The top region of the MPC contains an open space between the underside of the MPC lid and the top of the fuel basket. In addition the top of the fuel basket contains mouseholes. For conservatism, the open space portion of the top plenum is neglected in the MPC thermal model. This results in added resistance to the thermosiphon action in the plenum region, which overstates the cladding temperatures in the thermal models.

~~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:~~

$$\cancel{K_z \frac{d^2T^*}{dz^2} = -\Delta K \frac{d^2T_o}{dz^2}}$$

~~Where, ΔK 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^2T_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 ΔT_{ax} is the fuel cell temperature rise and L is the active fuel length. Conservatively postulating a lower bound ΔT_{ax} of 200°F and L of 12 ft, d^2T_o/dz^2 is computed as 1.39°F/ft^2. Integrating the perturbation equation shown above, the following formula for T^* is obtained:~~

$$\cancel{T^* = \left(\frac{\Delta K}{K_z}\right) \frac{d^2T_o}{dz^2} L^2}$$

~~Employing a conservative low value for the ($\Delta K/K_z$) parameter of 0.15, T^* is computed as 30°F. In other words, the baseline HI-STORM solution over predicts the peak cladding temperature by approximately 30°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 (~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. ~~Furthermore, heat dissipation by the aluminum heat conduction elements, if used, is conservatively neglected in the thermosiphon models employed in the HI-STORM modeling.~~

In previous revisions of this FSAR, the downcomer area was grossly understated in the FLUENT models. In Revision 2 of the FSAR, the downcomer area is still slightly understated for all MPC geometries (see table below), but the extent of conservatism has been moderated.

Comparison of the Actual and Assumed MPCs Downcomer Flow Areas			
	Actual (Based on drawings provided in Section 1.5)	*Assumed in the FLUENT Model (Revision 1)*	*Assumed in the FLUENT Model (Revision 2)*
MPC-24	*700.6*	*517.1*	*641.4*
MPC-24E & MPC-24EF	*664.9*	*517.1*	*641.4*
MPC-32 & MPC-32F	*773.3*	*517.1*	*746.1*
MPC-68, MPC-68F & MPC-68FF	*629.9*	*370.6*	*601.1*

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 elements of heat transfer (C3 and C5) are~~ *C3 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), *and conduction through the overpack structure.* ~~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 approac*h* 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:~~

~~$$T_{max} = T_{sink} + \Delta T_{air} + \Delta T_s$$~~

~~where, T_{sink} = Sink temperature (mean of inlet and outlet air temperature)~~
~~ΔT_{air} = Solid surface to air temperature difference~~
~~ΔT_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 ΔTs 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~~ *QQUANTIFICATION OF MARGINS~~THER~~* ~~CONSERVATISMS~~

Section 4.4.6 of the FSAR lists ~~eleven elements~~ *an array* of conservatism*s*, of which certain non-transparent and individually significant items are discussed in detail in this appendix. *To quantify individual thermal margins, several FLUENT sensitivity runs are performed using the limiting canister (MPC-68). In these runs each modeling parameter described below is individually changed from its original design value to a best estimate value and the Peak Cladding Temperature (PCT) re-computed. Finally, a composite thermal model is constructed with the modeling parameters set at their best estimate value and the combined PCT margin is obtained.*

1) Flow Resistance Margin

The axial flow resistance of the fuel cells is simulated using the porous media model. An explicit

3D evaluation has determined that the porous media model used in the HI-STORM thermal models over-predicts fuel cell pressure drop by 45%. To quantify the thermal margin, the porous the pressure drop overprediction is removed and peak clad temperature re-computed.

2) Fuel Basket Axial Conductivity Margin

In computing the axial conductivity of the fuel basket, the thermal design neglected the conductivity of fuel pellets. To quantify the thermal margin, the fuel pellet conductivities data reported in HI-STORM FSAR Table 4.2.3 are used to re-compute the MPC-68 fuel basket axial conductivities reported next:

Temperature (°F)	Design Axial Conductivity (Btu/ft-hr-°F)	Re-Computed Axial Conductivity (Btu/ft-hr-°F)
200	2.186	3.308
450	2.379	3.525
700	2.548	3.537

Using the re-computed axial conductivities the MPC-68 thermal model is run and peak cladding temperature is obtained.

3) Helium Thermal Conductivity Margin

The HI-STORM thermal calculations employ thermal conductivity of helium at 1 atm pressure. This assumption ignores the enhancement of helium thermal conductivity at elevated pressures within the MPC (~7 atm). According to Reid et. al. thermal conductivity of gases rise with pressure as:*

$$K(P) = K(P_o)\,(1 + \theta/100\,(P - P_o))$$

Where, $K(P)$ is the gas conductivity at pressure P, $K(P_o)$ is the conductivity at reference pressure P_o and θ is the pressure enhancement factor (1% per bar). Using the above information, $K(P)$ is used to define the helium conductivity in the HI-STORM model and peak cladding temperature obtained.

4) Downcomer Area Margin

As stated previously in Section 4.B.6 the MPC annulus downcomer areas in the FLUENT thermal models are understated. To uncover the thermal margin, the model downcomer area understatement is removed by increasing the annular gap and peak clad temperatures re-computed.

Finally, it is noted that the computed peak cladding temperatures reported in the HI-STORM

* *"The Properties of Gases and Liquids," 4th Edition, by R.C. Reid, J.M. Prausnitz and B.E. Poling, McGraw Hill, Inc., (1987).*

FSAR (Tables 4.4.9, 4.4.10 and 4.4.26) are lower than the 752°F temperature limit for all MPCs. This can be viewed as an additional thermal margin in the HI-STORM System. The computed margins provided in the FSAR are reproduced in Table 4-5.1.

Table 4.B.2 contains the results of thermal analysis wherein each of the four parameters (items (1) though (4)) discussed above are individually set to best-estimate values, and an additional run to obtain their composite effect. As Table 4-5.2 shows, the margin between the computed peak cladding temperature with all four parameters set at realistic values and the 752°F limit varies from a lower bound of 70°F for MPC-68, of which 21°F is due to the margin reported in the FSAR using the licensing basis model. Because MPC-32 and MPC-24 have much greater margins (90°F and 64°F, respectively) predicted by the licensing basis model, the total margin for these PWR MPCs will be accordingly larger (approximately 113°F to 139°F).

Table 4.B.1

Licensing Basis Model PCT Temperatures Margin From Limit

MPC Model	*Margin in the Peak Cladding Temperature*
MPC-24	*64*
MPC-32	*90*
MPC-68	*21*

Table 4.B.2

Lower Bound Individual Peak Clad Temperature Margins in the HI-STORM System

Item	*Reduction in the Computed Peak Cladding Temperature (°F)*
1. Flow Resistance	*36.5*
2. Fuel Basket Axial Conductivity	*8.0*
3. Helium Conductivity	*6.0*
4. Downcomer Area	*2.8*
A. Composite analysis with all four parameters (Items 1 through 4 above) set at realistic values	*49*
B. Margin from the limit (Table 4.B.1)	*21*
Total margin (A & B above)	*70*

~~*These conservatisms are primarily intrinsic to the modeling methodology or are product of assumptions in the input data. Examples in the latter category are values assumed in the thermal analysis for key inputs such as insolation heat, ambient temperature and emissivity of the heat exchange surfaces. Conservatisms in the former category includes implicit assumptions to under represent heat transfer, an example of this being the assumption that fuel pellets do not contribute to axial heat dissipation in the MPC. A listing of conservatisms is provided below:*~~

~~*Axial heat transfer through fuel pellets is neglected.*~~
~~*The upflow of helium through the MPCs is assumed to be laminar (high flow resistance, low heat transfer).*~~
~~*Heat dissipation by grid spacers, top & bottom fittings is ignored.*~~
~~*Insolation heating assumed with a bounding absorbtivity of 1.0.*~~
~~*Contact between fuel and basket and between basket and supports neglected.*~~
~~*MPC is assumed to be loaded with the most thermally resistive fuel type in its category (BWR or PWR) as applicable.*~~

~~*Finally, it should be noted that the computed peak cladding temperatures for all MPCs are also lower than the 400°C limit by varying amounts, which can be viewed as an additional thermal margin in the system. The assumptions inherent to the FLUENT solution methodology and to the solution process, in conjunction with those in the input data, are estimated to have an aggregate effect of overestimating cladding temperatures by a considerable amount, as estimated in Table 4.B.1.*~~

~~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.~~

~~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.~~

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† 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

† 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 heat load of the fuel, moisture is removed from the MPC using either a vacuum drying system or forced helium dehydration system. Section 4.5 of the FSAR has guidance on moisture removal requirements for the various heat loads. For lower heat loads, t~~T~~he 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~~*F*or higher ~~burn up fuel~~ *heat loads, or as an alternative for lower heat loads*, 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.~~

Limitations for the at-vacuum duration are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for requirements on moisture removal based on the heat load 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 heat load is greater than threshold limits from Section 4.5, a Supplemental Cooling System (SCS) 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[3] 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[1] 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[2]. The MPC may be lowered using the MPC downloader, the main crane hook or other similar devices. The MPC

1 Vent duct shield inserts are only used on the HI-STORM 100.

2 The alignment guide may be configured in many different ways to accommodate the specific sites. See Table 8.1.6.

3 Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid

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 (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

Note:
An underwater camera or other suitable viewing device may be used for monitoring underwater operations.
Note:
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.

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.

~~x.~~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 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:

Caution:
Limitations for the at-vacuum duration are evaluated and established on a canister-specific basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.

Note:
Vacuum drying or *forced helium dehydration* ~~moisture removal~~ (for higher ~~burn-up~~*heat load* 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 ~~8.1.22~~*6*i 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 6.1 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 though the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.

~~Note~~Caution:

Limitations for the handling of the loaded MPC in 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 specific time limits for transporting a loaded MPC in the HI-TRAC transfer CASK based on MPC heat loads. For MPCs containing greater than threshold heat loads, the Supplemental Cooling System (SCS) ~~must~~shall be used to prevent fuel cladding temperatures which exceed ISG-11 Rev. 3 limits. The implementation of cooling using the SCS must 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 operation. Use of the SCS shall continue until the MPC is ready for transfer to the HI-STORM.

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 as detailed in Section 4.5 of the FSAR.

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 4.4.37.*

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

ALARA Warning:
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.
Caution:
Limitations for the handling of the loaded MPC in 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 more detailed guidance on time limits based on MPC heat loads.

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,* ~~A~~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:

Warning:
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 heat loads requiring supplemental cooling, the time to complete the transfer is 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† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	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)††	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid) ††	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.

† See Table 8.1.2 for a description of each load handling case.
†† 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 COMPONENTS100-TON HI-TRAC**

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	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)††	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid)††	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.

† See Table 8.1.4 for a description of each load handling case.
†† 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†	Torque (ft-lbs) ††	Pattern†††
HI-TRAC Top Lid Bolts†	Hand tight	None
HI-TRAC Pool Lid Bolts (36 Bolt Lid)†	58 ft-lbs	Figure 8.1.37
HI-TRAC Pool Lid Bolts (16 Bolt Lid)†	110 ft-lbs	Figure 8.1.37
100-Ton HI-TRAC Transfer Lid Bolts†	203 ft-lbs	Figure 8.1.37
125-Ton HI-TRAC Transfer Lid Bolts†	270 ft-lbs	Figure 8.1.37
MPC Lift Cleats Stud Nuts†	793 ft-lbs	None
MPC Lift Hole Plugs†	Hand tight	None
Threaded Fuel Spacers	Hand Tight	None
HI-STORM Lid Nuts †	100 ft-lbs	None
HI-STORM 100S Lid Nuts† (Temporary and Permanent Lids)	Hand Tight +1/8 to ½ 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

† 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).

†† Unless specifically specified, torques have a +/- 5% tolerance.

††† No detorquing pattern is needed.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure[†]	Description
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.

[†] 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[†]	Description
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).

[†] 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[1]	Description
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 weds. 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.

[1] 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[†]	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.
Supplemental Cooling System~~MSLD (Helium Leakage Detector)~~	*~~Not Important To Safety~~Not Important to Safety*	*2.C.1~~Not shown~~*	*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 heat loads above threshold limits defined in Section 4.5.~~Used for helium leakage testing of the MPC closure welds.~~*
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.

[†] 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†

Instrument	Function
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	
~~Helium Mass Spectrometer Leak Detector (MSLD)~~*Deleted*	~~Ensures leakage rates of welds are within acceptance criteria.~~
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.

† 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.

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 heat load is greater than threshold limits from Section 4.5, a Supplemental Cooling System (SCS) is connected to the HI-TRAC annulus following transfer from the HI-STORM to the HI-TRAC and is 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[1] (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

[1] Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is 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 of the loaded MPC in 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 specific time limits for transporting a loaded MPC in the HI-TRAC transfer CASK based on MPC heat loads. Use of the Supplemental Cooling System (SCS) is required to maintain fuel cladding temperatures below limits when MPCs with greater than threshold limits are transported in the HI-TRAC. Operation of the SCS 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 operations.

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 in Section 4.5 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 preformed 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 preformed 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, p*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 LCO 3.3.1 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.~~

~~l.~~m. Remove the MPC lid-to-shell weld using the weld removal system. See Figure 8.1.9 for rigging.

~~m.~~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.

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 5×10^{-6} atm cm^3/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[†] 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.

[†] 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}B content (areal density)~~at~~ *in samples taken from* the ends of the panel. The minimum ^{10}B loading of the *neutron absorber*~~Boral~~ panels for each MPC model is provided in Table 2.1.15. Any panel in which ^{10}B loading is less than the minimum allowed shall be rejected. Testing shall be performed using written and approved procedures. Results shall be documented and become part of the cask quality records documentation package.

9.1.5.3.2 METAMIC® (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®, the following criteria must be satisfied:

- ***The boron carbide powder used in the manufacturing of METAMIC® must have small particle sizes to preclude neutron streaming***
- ***The ^{10}B areal density must comply with the limits of Table 2.1.15.***
- ***The B_4C powder must be uniformly dispersed locally, i.e. must not show any particle agglomeration. This precludes neutron streaming.***
- ***The B_4C powder must be uniformly dispersed macroscopically, i.e. must have a consistent concentration throughout the entire neutron absorber panel.***
- ***The maximum B_4C content in METAMIC® shall be less than or equal to 33.0 weight percent.***

To ensure that the above requirements are met the following tests shall be performed:

- ***All lots of boron carbide powder are analyzed to meet particle size distribution requirements.***
- ***The following qualification testing shall be performed on the first production run of METAMIC® panels for the MPCs in order to validate the acceptability and consistency of the manufacturing process and verify the acceptability of the METAMIC® panels for neutron absorbing capabilities:***

  1) The boron carbide powder weight percent shall be verified by testing a sample from forty different mixed batches. (A mixed batch is defined as a single mixture of aluminum powder and boron carbide powder used to make one or more billets. Each billet will produce several panels.) The samples shall be drawn from the mixing containers after

mixing operations have been completed. Testing shall be performed using the wet chemistry method.

2) *The ^{10}B areal density shall be verified by testing a sample from one panel from each of forty different mixed batches. The samples shall be drawn from areas contiguous to the manufactured panels of METAMIC® and shall be tested using the wet chemistry method. Alternatively, or in addition to the wet chemistry tests, neutron attenuation tests on the samples may be performed to quantify the actual ^{10}B areal density.*

3) *To verify the local uniformity of the boron particle dispersal, neutron attenuation measurements of random test coupons shall be performed. These test coupons may come from the production run or from pre-production trial runs.*

4) *To verify the macroscopic uniformity of the boron particle distribution, test samples shall be taken from the sides of one panel from five different mixed batches before the panels are cut to their final sizes. The sample locations shall be chosen to be representative of the final product. Wet chemistry or neutron attenuation shall be performed on each of the samples.*

- *During production runs, testing of mixed batches shall be performed on a statistical basis to verify the correct boron carbide weight percent is being mixed.*

- *During production runs, samples from random METAMIC® panels taken from areas contiguous to the manufactured panels shall be tested via wet chemistry to verify the ^{10}B areal density. This test shall be performed to verify the continued acceptability of the manufacturing process.*

The measurements of B_4C particle size, ^{10}B isotopic assay, uniformity of B_4C distribution and ^{10}B areal density shall be made using written and approved procedures. Results shall be documented.

9.1.5.3.3 *Installation of the Neutron Absorber Panels*

Installation of *neutron absorber*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)	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. b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements. 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. d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. e) Cleanliness of the MPC shall be verified upon completion of fabrication. f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.	a) The MPC shall be visually inspected prior to placement in service at the licensee's facility. b) MPC protection at the licensee's facility shall be verified. c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.	a) None.

(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† (~~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
Section 8.1.4							
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
Section 8.1.5							
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

† See notes at bottom of Table 10.3.4.

CHAPTER 11†: 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‡

† 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).

‡ *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 rupture*d* 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*~~ *106.6* psig in Table 4.4.14 at an average temperature of ~~*513.6*~~ *522.8*°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~~.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(106.6\,\text{psig} + 14.7)\,(522.8°\text{K} + 11.1°\text{K})}{522.8°\text{K}}$$

$$P_2 = 123.9\,\text{psia or } 109.2\,\text{psig}$$

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 ≤32°F and prohibits use of the HI-TRAC transfer cask below 0°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°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°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°F. The maximum off-normal environmental temperature is 100°F, which is an increase of 20°F. ~~Including the effect of a hypothetical 10% rod rupture condition on the MPC cavity gas conductivity, c~~*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°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°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°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is 0°F and the HI-TRAC is conservatively assumed to reach 0°F throughout the structure. For ambient temperatures from 0° to 32°F, ~~a 25%~~

~~ethylene glycol solution is~~*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.*3*~~4~~ 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.*4*~~5~~ 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~~ *18*°F, for the *fuel cladding. The results for the HI-STORM System are provided in Table 11.1.2.*~~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~~*18*°F, for the *fuel cladding.*~~MPC shell~~ *This is bounded by the 20°F temperature rise in Subsection 11.1.1, so the* ~~A bounding~~MPC internal pressure *calculated therein bounds the partial vents blockage condition as well.*
~~as a result of this calculated temperature increase is computed, based on initial conditions presented previously in Subsection 11.1.1.3, as follows:~~

$$\cancel{P_2 = P_1 \frac{T_1 + \Delta T}{T_1}}$$

~~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)~~

~~ΔT = Bounding MPC Temperature Rise (25°F or 19.9°K)~~

~~Substituting these values into the equation above, the bounding MPC internal pressure is obtained as:~~

$$\cancel{P_2 = 89.7 \times \frac{513.6 + 13.9}{513.6} = 92.1 psia = 77.4 psig}$$

~~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 ~~I~~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,~~. ~~a~~*A*lternatively, 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 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.

---

§ The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.

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

Location	Temperature [°F]	~~Design Basis Limits [°F]~~
Fuel Cladding	*751* ~~*711* (PWR)~~ ~~*760* (BWR)~~	~~1058 short-term~~
MPC Basket	~~*740*~~*726*	~~950 short-term~~
MPC Shell	~~*371*~~ *490*	~~775 short-term~~
~~Overpack Air Outlet~~	~~*226*~~	~~N/A~~
Overpack Inner Shell	~~*219*~~ *263*	~~350 short-term (overpack concrete)~~
Overpack Outer Shell	~~*165*~~ *203*	~~350 short-term (overpack concrete)~~

Table 11.1.2

BOUNDING[†] 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*~~ *731*	~~*765*~~ *749*	~~*821*~~	~~1058 short-term~~
MPC Basket	~~*720*~~ *706*	~~*745*~~ *723*	~~*801*~~	~~950 short-term~~
MPC Shell	~~*351*~~ *470*	~~*376*~~ *486*	~~*432*~~	~~775 short-term~~
Overpack Air Outlet *(mass flow averaged)*	~~*206*~~ *210*	~~*231*~~ *214*	~~*287*~~	~~N/A~~
Overpack Inner Shell	~~*199*~~ *243*	~~*224*~~ *266*	~~*280*~~	~~*400*~~ ~~350 short-term (overpack concrete)~~
Overpack Outer Shell	~~*145*~~ *177*	~~*170*~~ *182*	~~*226*~~	~~*600*~~ ~~350 short-term (overpack concrete)~~

~~† 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.~~

Table 11.1.3

STEADY-STATE MAXIMUM FUEL CLADDING TEMPERATURES FOLLOWING AN FHD FAILURE

MPC	*Design Heat Load (kW)*	*Computed Peak Clad Temp. (°F)*	*Off-Normal Temperature Limit (°F)*
MPC24/24E	*38*	*963*	*1058*
MPC-32	*38*	*926*	*1058*
MPC-68	*35.5*	*1017*	*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~~ *are not* 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 affect 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^2×°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^2 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~~*243*°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^2-hr)
h_{fc} = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft^2-hr-°F)
T_A = Fire Condition Temperature (1475°F)
T_S = Transient Surface Temperature (°F)
ε = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [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^2-hr)
h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_S = Transient Surface Temperature (°F)
T_A = Ambient Temperature (°F)
ε = 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^2-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 (2.6×10^5) 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, 1~~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)
ρ = Overpack Density (lb/ft^3)
L_c = Overpack Characteristic Length (ft)
k = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (142 lb/ft^3) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 127.7 hrs$$

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~~ *38* kW (~~98,090~~ *1.3×10^5* Btu/hr) for the 3.622 minute (0.0604 hours) fire duration results in the contained fuel assemblies heating up by only:

$$\Delta T_{fuel} = \frac{1.3 \times 10^5 \times 0.0604}{4680 + 2240} = 1.1°F$$

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 ~~5.5095~~*5.56*°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.~~636.8~~°F. Because the cladding temperature at the start of fire is substantially below the accident temperature limit (approximately 300°F lower)* the fuel cladding *temperature limit is not exceeded.* ~~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 11.2.1.2. 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*~~reported~~ in Subsection 11.2.1.2. 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 cruds.*~~

~~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 *and 4.4.26*~~, 4.4.46 and 4.4.27~~) and the short-term fuel cladding temperature limit (*Table 2.2.3*~~1058°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.~~ *The MPC-68 design yields the highest normal storage condition fuel cladding temperature. This design has the smallest margin to the accident temperature limit and consequently will have the shortest time to reach this limit and is, therefore, used in performing this evaluation.*

~~In the first step of the transient solution, t~~The decay heat load is set equal to *the design basis maximum value (Table 4.4.39), the inlets are blocked, and a transient solution is performed.*~~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), *fuel cladding, and all MPC and overpack steel component* temperatures remains below the*ir respective* short-term temperature limits through ~~72~~ *24* hours of *continuous full* 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 *maximum* temperatures at several points in the HI-STORM System at ~~33~~*24* 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, ~~reestablished,~~ 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 *bounding* pressure is *97.9 psig* ~~174.8~~ *(112.6* psia*)*, as reported in Table 4.4.14, at an average temperature of ~~513.6~~*528.0*°K. Using this pressure, a*n* ~~bounding~~ increase in the MPC cavity *bulk* temperature of 184°F (102.2°K, ~~maximum of MPC shell or fuel cladding~~*MPC cavity bulk* temperature rise ~~33~~*24* 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.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(112.6\ \text{psia})\,(528.0°\text{K} + 102.2°\text{K})}{528.0°\text{K}}$$

$$P_2 = 134.4\ \text{psia or } 119.7\ \text{psig}$$

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~~*HI-STORM System components and contents* temperatures approach their short-term temperature limits. At the specified time limit, ~~both the concrete section average and peak fuel cladding~~*all components and contents* 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 *cask system components and contents*~~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~~ *24* 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 *cask system components and contents*~~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 the*ir* short term temperature limit*s*, 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:

Δt = Allowable Burial Time (hrs)
m = Mass of HI-STORM System (lb)
c_p = Specific Heat Capacity (Btu/lb×°F)
ΔT = 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~~*34* 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 in-operable 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, a lower bound on the available time τ_{min} for fuel to remain at or below the limit (1058°F), is obtained under the following set of bounding assumptions:

1) *Design basis heat load*
2) *Instantaneous replacement of annulus water by air*

The analysis results for the peak cladding temperature as a function of time are plotted in Figure 11.2.8. From this plot τ_{min} is obtained as 33.4 hours. 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 2.2.3) are not exceeded, a bounding gas pressure is computed assuming fuel heatup from normal temperatures (Tables 4.4.9, 4.4.10 and 4.4.26) to a clad temperature limit (1058°F). For conservatism, the MPC average gas temperature is assumed to elevated from normal conditions to a 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

Material/Component	Initial† Condition (°F)	During Fire (°F)	Post-Fire†† Cooldown (°F)
Fuel Cladding	~~691 (MPC-24)~~ ~~691 (MPC-24E)~~ ~~691 (MPC-32)~~ ~~740 (MPC-68)~~*731*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~*732*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~*732*
MPC Fuel Basket	~~650 (MPC-24)~~ ~~650 (MPC-24E)~~ ~~660 (MPC-32)~~ ~~720 (MPC-68)~~*706*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~*707*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~*707*
Overpack Inner Shell	~~195~~*250*	300	~~195~~*300*
~~Overpack Radial Concrete Inner Surface~~	~~195~~	~~281~~	~~282~~
Overpack Radial Concrete Mid-Surface	~~173~~*222*	~~173~~*222*	~~184~~*235*
~~Overpack Radial Concrete Outer Surface~~	~~157~~	~~529~~	~~530~~
Overpack Outer Shell	~~157~~*200*	~~570~~*585*	~~570~~*585*

† Bounding ~~195~~*250*°F uniform inner surface and ~~157~~*200*°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_2 (Btu/lbm·°R)	0.056
Heat Capacity Lead (Btu/lbm·°R)	0.031
Maximum Decay Heat (kW)	~~28.7~~*438*
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[†]

Condition	Pressure (psig)			
	~~MPC-24~~	~~MPC-24E~~	~~MPC-32~~	~~MPC-68~~
~~Without Fuel Rod Rupture~~*Initial Condition*	~~79.8~~*107.4*	~~79.8~~	~~79.8~~	~~79.8~~
~~With 100% Fuel Rod Rupture~~*Bounding Maximum*	~~158.9~~*110.6*	~~159.3~~	~~191.1~~	~~126.6 (72.48-488)~~

~~[†] 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

Location	Initial Steady State Temperature [°F]	Bounding Temperature Rise [°F]	Hottest MPC Cross Section Peak Temperature [°F]
Fuel Cladding	~~872~~~~600~~~~455~~*745*	~~26.3~~*26.6*	~~898.3~~*771.6*
Basket Periphery	*620*	~~26.3~~*26.6*	~~626.3~~*646.6*
MPC Shell	*433*	~~26.3~~*26.6*	~~481.3~~*459.6*

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†
~~Bounding Decay Heat (kW)~~	~~28.74~~

† The cask is conservatively assumed to be at a uniform temperature *that conservatively bounds all normal storage temperatures.* ~~equal to the maximum fuel cladding temperature.~~

Table 11.2.7

MAXIMUM TEMPERATURES CAUSED BY EXTREME ENVIRONMENTAL TEMPERATURES[‡] [°F]

Location	Temperature	~~Accident Temperature Limit~~
Fuel Cladding	~~736 (PWR)~~ ~~785 (BWR)~~ *776*	~~1058~~
MPC Basket	~~765~~ *751*	~~950~~
MPC Shell	~~396~~*515*	~~775~~
Overpack Air Exit	~~251~~ *261*	~~N/A~~
Overpack Inner Shell	~~244~~ *288*	~~350 (overpack concrete)~~
Overpack Outer Shell	~~190~~ *228*	~~350 (overpack concrete)~~

~~‡ 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]

Temperature Location	Normal	Calculated Without Water in Water Jacket	~~Accident Condition Design Temperature~~
Fuel Cladding	~~872~~ *745*	~~888~~ *761*	~~1058 short-term~~
MPC Basket	~~852~~ *728*	~~868~~ *745*	~~950 short-term~~
MPC Basket Periphery	~~600~~ *620*	~~612~~ *630*	~~950 short-term~~
MPC Shell	~~455~~ *433*	~~466~~ *443*	~~775 short-term~~
~~HI-TRAC Inner Shell~~	~~322~~	~~342~~	~~400 long-term~~ ~~600 short-term~~
~~HI-TRAC Water Jacket Inner Surface~~	~~314~~	~~334~~	~~350 long-term~~
~~HI-TRAC Enclosure Shell Outer Surface~~	~~224~~	~~222~~	~~350 long-term~~
~~Axial Neutron Shield~~†	~~258~~	~~261~~	~~300 long-term~~

~~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.~~

~~† Local maximum section temperature.~~

Table 11.2.9

SUMMARY OF BLOCKED AIR INLET DUCT EVALUATION RESULTS

	Max. Initial Steady-State Temp.[‡] (°F)	Temperature Rise (°F)		Transient Temperature (°F)		~~Short-Term Temperature Limit (°F)~~
		at ~~33~~ *24* hrs	~~at 72 hrs~~	At *24* ~~33~~ hrs	~~at 72 hrs~~	
Fuel Cladding	~~740~~ *731*	~~101~~ *222*	~~160~~	~~841~~ *953*	~~900~~	~~1058~~
MPC Shell	~~351~~ *470*	~~184~~ *123*	~~250~~	~~535~~ *593*	~~601~~	~~775~~
Overpack Inner Shell ~~#1~~[‡‡]	~~199~~ *243*	~~113~~ *156*	~~174~~	~~312~~ *399*	~~373~~	~~600~~
~~Overpack Inner Shell #2~~[‡‡‡]	~~155~~	~~193~~	~~286~~	~~348~~	~~441~~	~~600~~
~~Overpack Outer Shell~~	~~145~~	~~14~~	~~40~~	~~159~~	~~185~~	~~600~~
Concrete Section Average	~~172~~ *183*	~~79~~ *53*	~~141~~	~~251~~ *236*	~~313~~	~~350~~

~~‡ Conservatively bounding temperatures reported includes a hypothetical rupture of 10% of the fuel rods.~~

~~‡‡ Coincident with location of initial maximum.~~

~~‡‡‡ Coincident with active fuel axial mid-height.~~

Table 11.2.10

MPC PRESSURES UNDER A POSTULATED FUEL HEATUP FROM NORMAL TEMPERATURES TO ACCIDENT LIMIT (1058°F)

MPC	*Normal Condition*		*Accident Pressure*[2]		*Design Pressure (From Chapter 2, Table 2.2.3)*
	MPC Average Temperature (T_o) [°F]	*Absolute Pressure (P_o) [psia] (Table 4.4.26)*	*Absolute (P) [psia]*	*Gage [psi]*	*Gage [psi]*
MPC-24	*479*	*112.6*	*182.0*	*167.3*	*200*
MPC-24E	*479*	*112.6*	*182.0*	*167.3*	*200*
MPC-32	*472*	*111.7*	*181.9*	*167.2*	*200*
MPC-68	*474*	*112.0*	*182.0*	*167.3*	*200*

2 Conservatively assuming the MPC is heated from T_o to a uniform maximum of 1058°F, the final gas pressure is computed by Ideal Gas Law as: $P = P_o (1058 + 460)/(T_o + 460)$.

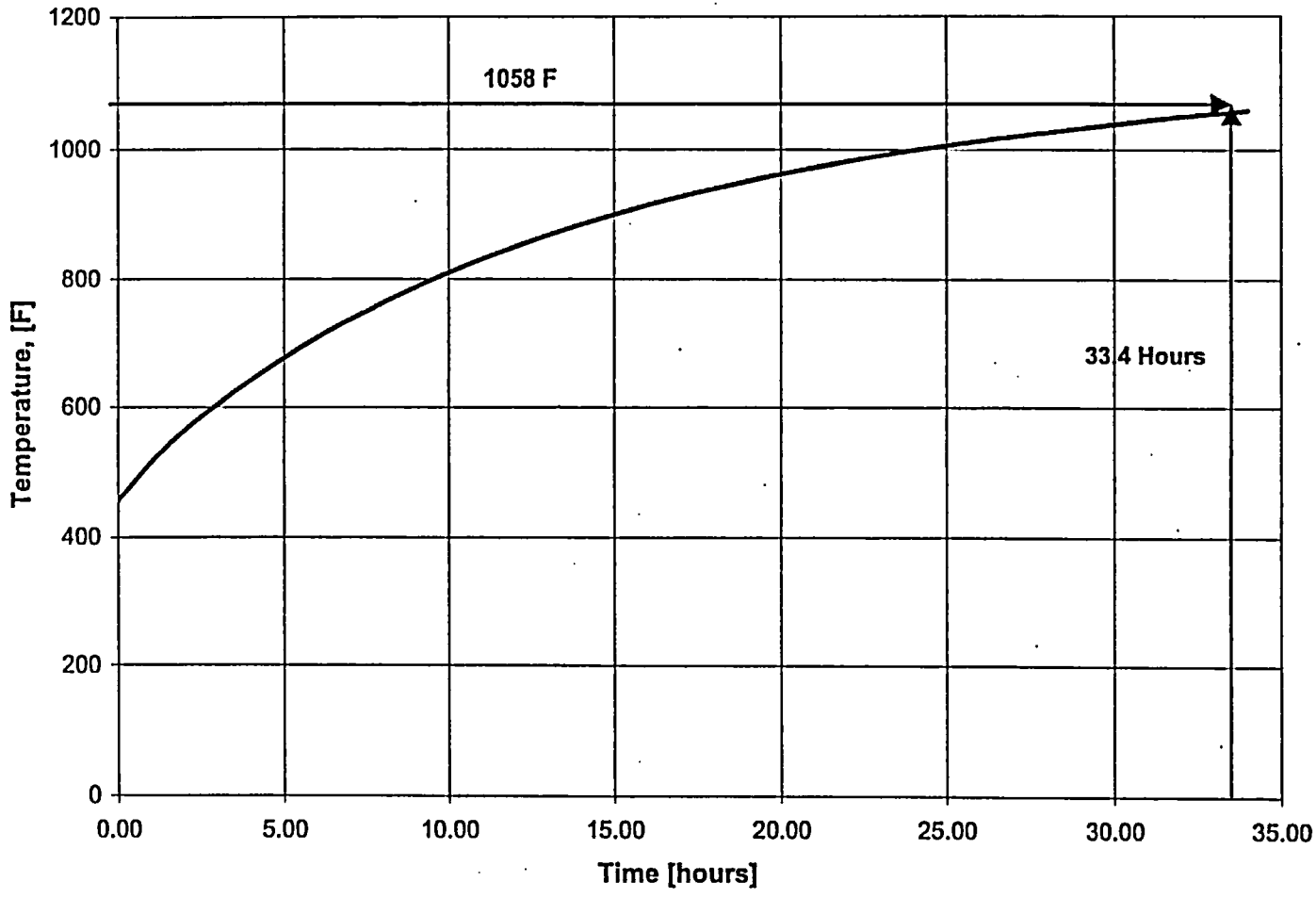

Figure 11.2.8: Peak Cladding Temperature in a Horizontal HI-TRAC

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†
Criticality Control	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) *3.1.4 Supplemental Cooling System* ~~5.6 Fuel Cladding Oxide Thickness Evaluation Program~~ ~~*(CoC 72-1014, Appendix B—Design Features)*~~
Shielding and Radiological Protection	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down 3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~ 3.2.2 TRANSFER CASK Surface Contamination ~~3.2.3OVERPACK Average Surface Dose Rates~~ *5.7 Radiation Protection Program*
Heat Removal Capability	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System *3.1.4 Supplemental Cooling System*
Structural Integrity	3.5 Cask Transfer Facility (CTF) ~~(CoC 72-1014, Appendix B—Design Features)~~ 5.5 Cask Transport Evaluation Program

† 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	*Supplemental Cooling System*
3.2.1	~~TRANSFER CASK Average Surface Dose Rates~~*Deleted*
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	~~OVERPACK Average Surface Dose Rates~~*Deleted*
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	ADMINSTRATIVE 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	~~Fuel Cladding Oxide Thickness Evaluation Program~~*Deleted*
5.7	*Radiation Protection Program*
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

The examples below execute 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.

Example 1

In this example a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern.

Step 1: Pick a value of X between 1 and 6. For this example X will be 2.8.

Step 2: Calculate $q_{Region\ 2}$ using equation 2.1.9.1:

$$q_{Region\ 2} = (2 \times 38)/[(1 + (2.8)^{0.15}) \times ((12 \times 2.8) + 20)] = 0.6543\ kW^{\dagger}$$

Step 3: Calculate $q_{Region\ 1}$ using equation 2.1.9.2:

† *This result is arbitrarily rounded to four decimal places.*

$q_{Region\ 1} = X \times q_{Region\ 2} = 2.8 \times 0.6543 = 1.83204\ kW$

Step 4: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 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.% ^{235}U or less to determine this fuel assembly's suitability for loading in this MPC-32.

Fuel Assembly Number 2 is not acceptable for loading unless a unique maximum allowable burnup for a cooling time of 3.3 years is calculated by linear interpolation between the values in Table 12.2.1 for 3 years and 4 years of cooling. Linear interpolation yields a maximum burnup of 43,597 MWD/MTU (rounded down from 43,597.9), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.

Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment that those in Table 12.2.1, which is still below the actual initial enrichment of Fuel Assembly Number 3. Due to its relatively low total decay heat of 0.5 kW (fuel: 0.4, non-fuel hardware: 0.1), Fuel Assembly Number 3 may be stored in Region 1 or Region 2.

Example 2

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern. Assuming the same value 'X', the same maximum fuel storage location decay heats are calculated. Equation 2.1.9.3 is executed for each fuel assembly using its exact initial enrichment to determine its maximum allowable burnup. Linear interpolation is used to further refine the maximum allowable burnup value between cooling times, if necessary.

Fuel Assembly Number 1: The calculated allowable burnup for 3.0 wt.% ^{235}U and a decay heat value of 1.83204 kW ($q_{region1}$) is 52,637 MWD/MTU at 4 years minimum cooling. Its decay heat is too high for loading in Region 2. Comparing the fuel assembly burnup and total decay heat of the contents[†] (fuel (1.01 kW) plus non-fuel hardware (0.5 kW)) to the calculated limits indicates that the fuel assembly, including the non-fuel hardware, is acceptable for storage in Region 1.

[†] *The assumption is made that the non-fuel hardware meets the burnup and cooling time limits in Table 2.1.25.*

Fuel Assembly Number 2: The calculated allowable burnup for 3.2 wt.% ^{235}U and a decay heat value of 1.83204 kW ($q_{region1}$) is 39,844 MWD/MTU for 3 years cooling and 53,195 MWD/MTU for 4 years cooling. Linearly interpolating between these values for a cooling time of 3.3 years yields a maximum allowable burnup of 43,849 MWD/MTU and, therefore, the assembly is acceptable for storage in Region 1. This fuel assembly's decay heat is also too high for loading in Region 2.

Fuel Assembly Number 3: The calculated allowable maximum burnup for 4.3 wt.% ^{235}U and a decay value of 0.6543 ($q_{region2}$) is 48,891 MWD/MTU for 18 years cooling. Comparing the fuel assembly burnup and total decay heat of the contents (fuel plus non-fuel hardware) against the calculated limits indicates that the fuel assembly and non-fuel hardware are acceptable for storage. Therefore, the assembly is acceptable for storage in Region 2. This fuel assembly would also be acceptable for loading in Region 1 (this conclusion is inferred, but not demonstrated).

Table 12.2.1

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment = 3.1 wt.% ^{235}U)
($q_{Region\ 1}$ = 1.83204 kW, $q_{Region\ 2}$ = 0.6543 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	39604	13568
≥4	52917	21040
≥5	61912	27028
≥6	68200	31446
≥7	68200	34520
≥8	68200	36814
≥9	68200	38588
≥10	68200	40059
≥11	68200	41280
≥12	68200	42375
≥13	68200	43349
≥14	68200	44263
≥15	68200	45144
≥16	68200	45991
≥17	68200	46809
≥18	68200	47628
≥19	68200	48433
≥20	68200	49247

Table 12.2.2

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment =4.185 wt.% ^{235}U)
($q_{Region\ 1}$ = 1.83204 kW, $q_{Region\ 2}$ = 0.6543 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	42060	14019
≥4	55800	21814
≥5	65028	27883
≥6	68200	32304
≥7	68200	35386
≥8	68200	37695
≥9	68200	39482
≥10	68200	40989
≥11	68200	42247
≥12	68200	43365
≥13	68200	44377
≥14	68200	45326
≥15	68200	46237
≥16	68200	47115
≥17	68200	47947
≥18	68200	48796
≥19	68200	49629
≥20	68200	50466

Table 12.2.3

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 16x16A)

FUEL ASSEMBLY NUMBER	ENRICHMENT (wt. % ^{235}U)	FUEL ASSEMBLY BURNUP (MWD/MTU)	FUEL ASSEMBLY COOLING TIME (years)	FUEL ASSEMBLY DECAY HEAT (kW)	NON-FUEL HARDWARE STORED WITH ASSEMBLY	NFH DECAY HEAT (kW)
1	3.0	37100	4.7	1.01	BPRA	0.5
2	3.2	40250	3.3	1.75	NA	NA
3	4.3	41976	18.2	0.4	BPRA	0.1

BASES TABLE OF CONTENTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY ..... B 3.0-1
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY ..... B 3.0-5

3.1 SFSC INTEGRITY ..... B 3.1.1-1
3.1.1 Multi-Purpose Canister (MPC) .....B 3.1.1-1
3.1.2 SFSC Heat Removal System ..... B 3.1.2-1
3.1.3 Fuel Cool-Down ..... B 3.1.3-1
3.1.4 Supplemental Cooling System.....B 3.1.4-1

3.2 SFSC RADIATION PROTECTION ..... B 3.2.1-1
3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~*Deleted* .B 3.2.1-1
3.2.2 TRANSFER CASK Surface Contamination ..... B 3.2.2-1
3.2.3 ~~OVERPACK Average Surface Dose Rates~~*Deleted* ..... B 3.2.3-1

3.3 SFSC CRITICALITY CONTROL .....B 3.3.1-1
Boron Concentration .....B 3.3.1-1

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES

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.

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, *performed for design-basis heat load,* 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)

BASES

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. *The results of this event are dependent upon the decay heat load of the MPC. Therefore, two analyses were performed. The first analysis was performed assuming a decay heat load of 28.74 kW at the Amendment 1 helium fill pressure.* None of the components reach their temperature limits over the 72-hour duration of the analyzed event. *The second analysis was performed for the MPC-68 at its design decay heat load of 35.5 kW and the Amendment 2 helium fill pressure. All component temperatures remain below their respective short term temperature limits up to 24 hours after event initiation. The MPC-68 analysis provides a bounding case for all MPC models since it yields the highest component temperatures.* ~~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)

BASES

LCO (continued)

This LCO is not intended to address low frequency, unexpected Design Event III and IV class events such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.9 of Appendix B to the CoC.

APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate ~~heat transfer~~*dissipation* of the decay heat ~~away~~ from the fuel assemblies.

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)

BASES

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* 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 *if the decay heat load of the MPC is less than or equal to 28.74 kW or within 16 hours if the decay heat load of the MPC is greater than 28.74 kW. These Completion Times are consistent with the two thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 72 or 24 hours after event initiation, for MPC heat loads of 28.74 and the highest authorized heat load of* ~~41.22~~ *38 kW, respectively. The 35.5 kW case analyzed for MPC-68 bounds the 38 kW case for the PWR MPCs.*

(continued)

BASES

ACTIONS

B.2.1 (continued)

The *two* Completion Times reflects *the 8 hours to complete Required Action A.1 and the appropriate balance of time consistent with the applicable analysis results. In each case, 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 unforseen 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)

BASES

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 Times of ~~48~~ *64 and 16* hours reflects *the Completion Times 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.~~

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)

BASES

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.

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 3.1.4 Supplemental Cooling System

BASES

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.

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 (≤ 45,000 MWD/MTU), SCS operation is only required during onsite transportation if the heat load is above the analyzed value and the TRANSFER CASK is in the horizontal orientation, because the fuel cladding temperature could exceed limit of 1058°F for moderate burnup fuel (Refs. 2 and 3). The TRANSFER CASK is considered to be horizontal any time it is significantly off of the vertical position.

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 at higher heat loads in order to meet the lower fuel cladding temperature limit. In the vertical orientation, the MPC/TRANSFER CASK assemblage is able to reject heat at a higher rate than in the horizontal orientation. The thermal analysis has established certain "threshold" heat loads for each TRANSFER CASK orientation at or beyond which the SCS is required for MPCs containing one or more high burnup fuel assemblies.

(continued)

BASES

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. Due to differences in heat rejection capability, different requirements apply based on the physical orientation of the TRANSFER CASK.

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 either of the following sets of conditions are met:

Set 1 (Applicability 'a' and 'b' and c.i or c.ii): MPCs having one or more fuel assemblies with an average burnup greater than 45,000 MWD/MTU and a decay heat load that exceeds one of the following:

> 23 kW with the TRANSFER CASK in the vertical orientation,

or

> 19 kW with the TRANSFER CASK in the horizontal orientation

Set 2: (Applicability 'a' and 'd' and 'e' and 'f'): MPCs having all fuel assemblies with an average burnup less than or equal to 45,000 MWD/MTU, a decay heat load greater than 30 kW, and the TRANSFER CASK in the horizontal orientation.

(continued)

BASES

APPLICABILITY (continued)

The Applicability is modified by a note stating that, upon reaching steady state operation, the SCS may be temporarily disabled for a short duration (≤ 7 hours) in order to facilitate necessary operational evolutions, such as movement of the TRANSFER CASK through a door or other similar operation. This note is acceptable because the thermal analysis shows that the fuel cladding temperature will not exceed the 400°C (752°F) temperature limit at design basis heat load for over 7 hours, with no water in the TRANSFER CASK-to-MPC annulus. "Steady state" is defined as no significant change in the SCS water temperature exiting the TRANSFER CASK.

ACTIONS

A.1

If the SCS has been determined to be inoperable and the TRANSFER CASK is in the vertical orientation, 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)

BASES

ACTIONS (continued)

A.2.1 and A.2.2

Required Actions A.2.1 and A.2.2 apply to a TRANSFER CASK in the horizontal orientation. Required Action A.2.1 requires the TRANSFER CASK to be rotated to the vertical condition, which puts the TRASNFER CASK and MPC in an analyzed condition as described in Required Action A.1. The Completion Time of 24 hours for Required Action A.2.1 is based upon the thermal analysis of the horizontally oriented TRANSFER CASK with water in the annulus, which will be the case if the SCS fails. Required Action A.2.2 is identical to Required Action A.1 and ensures adequate fuel cooling for an indefinite period of time while actions are under way to restore the SCS to operable status.

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.

(continued)

BASES

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.

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*