APPENDIX 1.D: Requirements on HI-STORM 100 Shielding Concrete

1.D.1 Introduction

The HI-STORM 100 overpack utilizes plain concrete for neutron and gamma shielding. Plain concrete used in the HI-STORM overpack provides only a compressive strength structural function due to the fact that both the primary and secondary load bearing members of the overpack are made of carbon steel. While most of the shielding concrete used in the HI-STORM 100 overpack is installed in the annulus between the concentric structural shells, smaller quantities of concrete are also present in the pedestal shield and the overpack lid. Because plain concrete has little ability to withstand tensile stresses, but is competent in withstanding compressive and bearing loads, the design of the HI-STORM 100 overpack places no reliance on the tension-competence of the shielding concrete.

During normal operations of the HI-STORM, the stresses in the concrete continuum are negligible, arising solely from its self-weight. ACI 318.1-89(92) provides formulas for permissible compressive and bearing stresses in plain concrete, which incorporate a penalty over the corresponding permissible values in reinforced concrete. The formulas for permissible compressive and bearing stresses set forth in ACI 318.1-89(92) are used in calculations supporting this FSAR in load cases involving compression or bearing loads on the overpack concrete. However, since the overpack concrete is designated as an ITS Category B material, it is appropriate to ensure that all "*critical characteristics*" of the concrete, as defined herein, are fully satisfied. During normal storage operations, the overpack concrete is completely enclosed by the overpack steel structure, protecting it from the deleterious effects of direct exposure to the environment, typical of most concrete structures governed by the ACI codes.

The "*critical characteristics*" of the plain concrete in the HI-STORM overpack are: (i) its density and (ii) its compressive strength. This appendix provides the complete set of criteria applicable to the plain concrete in the HI-STORM 100 overpack.

1.D.2 Design Requirements

The primary function of the plain concrete is to provide neutron and gamma shielding. As plain concrete is a competent structural member in compression, the plain concrete's effect on the performance of the HI-STORM overpack under compression loadings is considered and modeled in the structural analyses, as necessary. The formulas for permissible compressive and bearing stresses set forth in ACI 318.1-89(92) are used. However, as plain concrete has very limited capabilities in tension, no tensile strength capability is allotted to the HI-STORM concrete.

The steel structure of the HI-STORM overpack provides the strength to meet all load combinations specified in Chapters 2 and 3, due to the fact that both the primary and secondary load bearing members (as defined in the ASME Code, Section III, Subsection NF-1215) of the HI-STORM overpack are made from carbon steel. Credit for the structural strength of the plain concrete is only taken to enhance the compressive load carrying capability of the concrete in calculations appropriate to handling and transfer operations, and to demonstrate that the HI-STORM 100 System continues to provide functional performance in a post-accident environment. Therefore, the load combinations provided in ACI 349 and NUREG-1536, Table 3-1 are not applicable to the plain concrete in the HI-STORM overpack.

The shielding performance of the plain concrete is maintained by ensuring that the minimum concrete density is met during construction and the allowable concrete temperature limits are not exceeded. The thermal analyses for normal and off-normal conditions demonstrate that the plain concrete does not exceed the allowable long term temperature limit provided in Table 1.D.1. The temperature limits for transient conditions (such as under partial or full duct blockage and fire) are set down in Holtec position paper [1.D.4] in conformance with the guidelines set forth in the ACI code, supplemented by data from the published permanent literature. In reference [1.D.4] the guidance from ACI and the published archival data on plain concrete has been used to establish temperature limits for normal and transient conditions listed in Table 1.D.1 herein.

1.D.2.1 Test Results to Support Normal Condition Temperature Limit

Note 3 to Table 1.D.1 references Paragraph A.4.3 of ACI-349, which requires that normal condition temperatures in excess of 150°F bulk and 200°F local must be supported by test data to demonstrate that strength reductions are acceptable and that concrete deterioration does not occur. Such data are described and discussed in this subsection.

With respect to concrete compressive strength at bulk temperatures up to 300°F, test studies for elevated temperatures were performed by Carette and Malhotra [1.D.1] that examined conditions very similar to those of the HI-STORM concrete. Their tests were performed on 4" diameter by 8" long test cylinders. The test conditions most closely match the HI-STORM concrete with respect to water-to-cement ratio, limestone aggregate and temperature. While the HI-STORM storage period is much greater than 4 months (testing duration per [1.D.1]), the investigators state "any major strength loss is found to occur within the first month of exposure." The four-month compressive strength for these conditions was actually determined to be greater than the nominal concrete strengths despite the elevated temperatures. This is attributable to the increase in compressive strength that accompanies concrete aging, which more than offsets the temperature effects. Per [1.D.1], approximately 45% reduction in compressive strength and about 5% reduction in weight is observed in limestone concrete (water to cement ratio, w/c = 0.33) exposed for one month at 300°C (572°F).

Exposure of concrete to elevated temperatures can affect its properties due to the dehydration or loss of absorbed and chemically combined water. With respect to concrete shielding performance at local temperatures above 300°F, [1.D.4] examined weight loss and thermal degradation mechanisms of concrete at elevated temperatures.

To address a postulated accident that may occur during the 30-day vent blockage condition (i.e. a tornado borne missile impact), the compressive strength of the concrete is conservatively reduced by 50% even though the maximum temperature experienced by concrete during 30-day vent blockage accident is less than 450°F and is about 300°F during long term normal condition. The evaluations (Supplements 15 and 25 of HI-2012769) conclude that the concrete in overpack, post 50% strength reduction, is acceptable during and after the 30-day vent blockage accident, and during the long term normal condition.

To evaluate the effect of hydrogen loss on the shielding performance of the HI-STORM 100, it was assumed that entire hydrogen is lost from the concrete. This is an excessively conservative estimate of an upper bound dose rate effect. Water and hydrogen is present in concrete in two forms, chemically bound water/hydrogen, and physically bound water. The material properties for the concrete in the HI-STORM 100 assume hydrogen content that is less than or maybe equal to that in the chemically bound water. As the entire weight loss is attributed to water loss, the total amount of hydrogen in concrete is expected to be no less than 0.8 wt%. The assumed hydrogen content in concrete composition listed in Table 5.3.2 of the FSAR is only 0.6 wt%, which is used for shielding analyses. The analyses results show that shielding performance with such reductions in hydrogen content is negligible. Additionally, the results demonstrate that the hypothetical HI-STORM 100% duct blockage accident condition is bounded by the HI-TRAC accident condition discussed in Section 5.1.2 of the FSAR.

1.D.3 Material Requirements

Table 1.D.1 provides the material limitations and requirements applicable to the overpack plain concrete. These requirements, drawn from ACI 349-85 and supplemented by the provisions of NUREG 1536 (page 3-21), are intended to ensure that the "*critical characteristics*" of the concrete placed in the HI-STORM overpack comply with the requirements of this Appendix and standard good practice. Two different minimum concrete densities are specified for the overpack concrete, based on the presence or absence of the steel shield shell. The steel shield shell was deleted from the overpack design after the construction of overpack serial number 1024-7.

ACI 349 was developed to govern the design and construction of steel reinforced concrete structures for the entire array of nuclear power plant applications, except for concrete reactor vessels and containment structures. Therefore, ACI 349 contains many requirements not germane to the plain concrete installed in and completely enclosed by the steel HI-STORM overpack structure. For example, the overpack concrete is not exposed to the environment, so provisions in the standard for protecting concrete from the environment would not be applicable to the concrete contained in the overpack.

In accordance with the requirement in Section 3.3 of Appendix B of the HI-STORM 100 CoC, Section 1.D.4, Table 1.D.1 and Table 1.D.2 were developed using the guidance of ACI 349-85, to the extent it needs to be applied to the unique application of placing unreinforced concrete inside the

[1.D.1] Carette and Malhotra, "Performance of Dolostone and Limestone Concretes at Sustained High Temperatures", Temperature Effects on Concrete, ASTM STP 858.

[1.D.2] Deleted

[1.D.3] Concrete Manual, 8th Edition, US Bureau of Proclamation, Denver, Colorado, 1975.

[1.D.4] Holtec Position Paper, DS-289, "Maximum Permissible Temperature in Plain Concrete in HI-STORM System Components Under Off-Normal and Accident Conditions", Revision 5.

Table 1.D.1
Requirements for Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density in overpack body (Minimum) (see Table 3.2.1 for information on maximum concrete density)	140 lb/ft^3
Density in lid and pedestal (Minimum) (See Table 3.2.1 for information on maximum concrete density)	140 lb/ft^3 (HI-STORM 100S Version B does not have a concrete-filled pedestal)
Specified Compressive Strength	3,300 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318.1-89(92)
Cement Type and Mill Test Report	Type II; (ASTM C 150 or ASTM C595)
Aggregate Type	Fine and coarse aggregate as required (Note 2)
Nominal Maximum Aggregate Size	1-1/2 (inch)
Water Quality	Deleted
Material Testing	See Note 4.
Admixtures	Deleted
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) (See Table 1.D.2, Note 1)
Concrete Quality	Deleted
Mixing and Placing	See Note 6.
Consolidation	Deleted
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Through-Thickness Section Average† Temperature Limit Under Long Term Conditions	300°F (See Note 3)
Maximum Local Temperature Limit Under 30-Day 100% Vent Blockage Accident Condition	(See Table 2.2.3)
Maximum Local Temperature Limit Under Short Term/Accident/Off Normal Conditions	(See Table 2.2.3)
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, marble, basalt, granite, gabbros, 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 overpack concrete.

Table 1.D.1 (continued)
Requirements for Plain Concrete

Notes:

1. Deleted

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 aggregates that have been shown by special tests or actual service to produce concrete of adequate strength, unit weight, and durability meeting the requirements of Tables 1.D.1 and 1.D.2 are acceptable in accordance with ACI 349 Section 3.3.2. The high-density coarse aggregate percentage of Material Finer than No. 200 Sieve may be increased to 10 % if the material is essentially free of clay or shale.

3. The 300°F long term temperature limit is specified in accordance with Paragraph A.4.3 of Appendix A to 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.

4. Tests of materials and concrete, as required, shall be made in accordance with standards of the American Society for Testing and Materials (ASTM) as specified here, to ensure that the *critical characteristics* for the HI-STORM concrete are achieved. ASTM Standards to be used include: C 31-96, C 33-82, C 39-96, C 88-76, C 131-81, C 138-92, C 143-98, C 150-97, C 172-90, C 192-95, C 494-92, C 637-73. More recent approved editions of the referenced standards may be used.

5. Deleted

6. Water and admixtures may be added at the job site to bring both the slump and wet unit weight of the concrete within the mix design limits. Water or admixtures shall not be added to the concrete after placement activities have started. The tolerance for individual and combined aggregate weights in the concrete batch may be outside of tolerances specified in ASTM C94, provided that the wet unit weight of the concrete is tested prior to placement and confirmed to be within the approved range.

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

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

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

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

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

<u>Thermal</u>

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 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel and 570ºC (1058ºF) for moderate burnup fuel.

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

and aggregate requirements to allow the utilization of the temperature limits in Table 2.2.3. The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in Table 2.2.3.

The overpack is designed for extreme cold conditions, as discussed in 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 to a real individual beyond the controlled area boundary is limited by 10CFR72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/year for other critical organs, including contributions from all nuclear fuel cycle operations. Since these limits are dependent on plant operations as well as site-specific conditions (e.g., the ISFSI design and proximity to the controlled area boundary, and the number and arrangement of loaded storage casks on the ISFSI pad), the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a single cask and a range of typical ISFSIs using the HI-STORM 100 System are provided in Chapter 5. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10CFR72.212.

The overpack is designed to limit the calculated surface dose rates on the cask for all 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.

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 Regulatory Guide 3.61 for non-redundant lifting devices.

The HI-TRAC transfer cask is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the HI-TRAC transfer cask must protect the MPC from deformation, provide continued adequate performance, and allow the retrieval of the MPC under all conditions. These design loadings include a side drop from the maximum allowable handling height, consistent with the technical specifications. The load combinations for which the HI-TRAC is designed are defined in Section 2.2.7. The physical characteristics of each MPC for which the HI-TRAC is designed are defined in Chapter 1.

Table 2.0.1
MPC DESIGN CRITERIA SUMMARY

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

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

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/3 Yield	NUREG-0612 Regulatory Guide 3.61	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
Fuel Rod Rupture Releases:			
Source Term Release Fraction	1%	NUREG-1536	Sections 2.2.1.3
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3
Snow and Ice	Protected by Overpack	ASCE 7-88	Section 2.2.1.6
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.2.2
Leakage of One Seal	N/A	ISG-18	Sections 2.2.2.4 and 7.1
Partial Blockage of Overpack Air Inlets	50% of Air Inlets Blocked	-	Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10%	NUREG-1536	Sections 2.2.2.1
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Tip Over	See Table 2.0.2	-	Section 2.2.3.2
End Drop	See Table 2.0.2	-	Section 2.2.3.1
Side Drop	See Table 2.0.3	-	Section 2.2.3.1
Fire	See Tables 2.0.2 and 2.0.3	10CFR72.122(c)	Section 2.2.3.3

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

Type	Criteria	Basis	FSAR Reference
Response and Degradation Limits	Protect MPC from deformation	10CFR72.122(b) 10CFR72.122(c)	Sections 2.0.2 and 3.1
	Continued adequate performance of overpack	10CFR72.122(b) 10CFR72.122(c)	
	Retrieval of MPC	10CFR72.122(l)	
Thermal:			
Maximum Design Temperatures:			
Concrete			
Through-Thickness Section Average (Normal)	Table 2.2.3	ACI 349, Appendix A (Paragraph A.4.3)	Section 2.0.2, and Tables 1.D.1 and 2.2.3
Through-Thickness Section Average (Off-Normal and Accident)	Table 2.2.3		Section 2.0.2, and Tables 1.D.1 and 2.2.3
Steel Structure (other than lid bottom and top plates) Lid Bottom and Top Plates	350° F 450°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	10CFR72.122(f) & (l)	Section 3.4
Accident			Section 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^{o} F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†:		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	250,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.4 Section 2.2.1.2
Dry Loaded MPC (max.)	90,000 lb.	R.G. 3.61	Table 3.2.1 and Section 2.2.1.2
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3.5
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature During Handling Operations	0^{o} 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

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

TABLE 2.0.3
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Section 2.0.3
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design Codes:			
Structural Steel	ASME Code, Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.3
Lifting Trunnions	NUREG-0612 & Regulatory Guide 3.61	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights†:			
Max. Empty Cask:			
w/top lid and pool lid installed and water jacket filled	143,500 lb. (HI-TRAC 125) 102,000 lb. (HI-TRAC 100) 102,000 lb. (HI-TRAC 100D) 146,000 lb. (HI-TRAC 125D)	R.G. 3.61	Table 3.2.2
w/top lid and transfer lid installed and water jacket filled (N/A for HI-TRAC 100D and 125D)	155,000 lb. (HI-TRAC 125) 111,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.2
Max. MPC/HI-TRAC with Yoke (in-pool lift):	250,000 lb. (HI-TRAC 125 and 125D) 200,000 lb. (HI-TRAC 100 and 100D)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3
Water Jacket Cavity	60 psig (internal)	ANSI/ANS 57.9	Section 2.2.1.3

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

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

Type	Criteria	Basis	FSAR Reference
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature:	80 °F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†			
Max. Loaded Canister			
Dry	90,000 lb.	R.G. 3.61	Table 3.2.1
Wet (including water in HI-TRAC annulus)	106,570 lb.	R.G. 3.61	Table 3.2.4
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/3 Yield	NUREG-0612 Regulatory Guide 3.61	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature for Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Test Loads:			
Trunnions	300% of vertical design load	NUREG-0612 & Regulatory Guide 3.61	Section 9.1.2.1
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1
Fire			
Duration	4.8 minutes	10CFR72.122(c)	Section 2.2.3.3

† 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.

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

2.1.2 Intact SNF Specifications

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

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

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

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in Table 1.0.1.

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

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 j

Where:

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

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

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

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

2.1.9.1.4 Other Considerations

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

- Calculated burnup limits shall be rounded down to the nearest integer

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

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

- ZR-clad fuel assemblies must have a minimum enrichment, as defined in Table 1.0.1, greater than or equal to the value used in determining the maximum allowable burnup per Section 2.1.9.1.3 to be authorized for storage in the MPC.

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

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

2.1.9.1.5 Supplemental Cooling Threshold Heat Loads

Fuel loading operations involving the handling of High Burnup Fuel (HBF) in a dewatered MPC emplaced in a HI-TRAC transfer cask require additional cooling under certain thermal loads to address reduced heat dissipation relative to the normal storage condition. To address this requirement the Supplemental Cooling System (SCS) defined in Appendix 2.C is mandated under threshold heat loads defined in Section 4.5 and Table 2.1.30. The specific design of a SCS must accord with site-specific needs and resources, including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the SCS are satisfied by plant-specific designs are set forth in Appendix 2.C.

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

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

Table 2.1.12

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

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

Table 2.1.16

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

Fuel Assembly Array/Class	**All Intact Fuel Assemblies**		**One or More Damaged Fuel Assemblies or Fuel Debris**	
	Max. Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G/I	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A	1,400	2,000	1,500	2,300
17x17A	1,600	2,200	1,800	2,600
17x17B/C	1,900	2,600	2,100	2,900

Note:

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

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

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

Table 2.1.20

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

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class	Uranium oxide PWR damaged fuel assemblies and/or fuel debris meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time, and Average Burnup per Assembly	ZR clad: 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	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including DFC and non-fuel hardware)

Table 2.1.22

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

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide or MOX BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels.	Uranium oxide or MOX BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, in DFCs.
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class
Maximum Initial Planar Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable fuel assembly array/class	Planar Average: ≤ 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-68 AND MPC-68FF

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

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

Table 2.1.24

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

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class	Uranium oxide, PWR damaged fuel assemblies and fuel debris in DFCs meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3	As specified in Table 2.1.3
Post-irradiation Cooling Time and Average Burnup per Assembly	*ZR clad:* 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	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including DFC and non-fuel hardware)

Table 2.1.25

NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, 3, and 7)

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

NOTES:

1. Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.
2. Linear interpolation between points is permitted, except that NSA or Guide Tube Hardware and APSR burnups > 180,000 MWD/MTU and $\leq$ 630,000 MWD/MTU must be cooled $\geq$ 14 years and $\geq$ 11 years, respectively.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), vibration suppressor insertsand Neutron Source Assemblies (NSAs) in combination with other control components (i.e. BPRAs, TPDs, and/or RCCAs).
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, orifice rod assemblies, Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), Rod Cluster Control Assemblies (RCCAs) and NSAs without other forms of control components.
6. N/A means not authorized for loading at this cooling time.
7. Non-fuel hardware burnup and cooling time limits are not applicable to Instrument Tube

Table 2.1.28
PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.28 (cont'd)

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

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

Table 2.1.29

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

Table 2.1.29 (cont'd)

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

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

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^{o}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^{o} F.

Special lifting devices shall meet the requirements of ANSI N14.6† [2.2.3].

2.2.1.3 Pressure

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

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

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

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

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

2.2.8 Allowable Stresses

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

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

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

S_y: Minimum yield strength at temperature

S_u: Minimum ultimate strength at temperature

2.2.9 Requirements on Lifting and Special Lifting Devices

2.2.9.1 Definitions: The lifting and handling systems used in Holtec's used fuel management program are made up of individual components or devices. These components can be

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further classified as either "*lifting devices*" or "*special lifting devices*." (See Glossary)

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

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

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

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

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

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

2.2.9.3 Single Failure Proof Criteria

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

should be doubled, or a redundant *lifting device* should be provided. The load drop protection feature incorporated in the vertical towers of a cask transporter is an example of redundant lifting part.

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

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

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

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

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Table 2.2.3

DESIGN TEMPERATURES

HI-STORM 100 Component	Long Term, Normal Condition Design Temperature Limits (Long-Term Events) (^{o} F)	Off-Normal and Accident Condition Temperature Limits (Short-Term Events)† (^{o} F)	30-Day Accident Condition Temperature Limit (^{o} F)*
MPC shell	500	775	572
MPC basket	725	950	752
MPC Neutron Absorber	800	1000	752
MPC lid	550	775	572
MPC closure ring	400	775	572
MPC baseplate	400	775	572
HI-TRAC inner shell	400	800	-
HI-TRAC pool lid/transfer lid	350	800	-
HI-TRAC top lid	400	800	-
HI-TRAC top flange	400	700	-
HI-TRAC pool lid seals	350	N/A	-
HI-TRAC bottom lid bolts	350	800	-
HI-TRAC bottom flange	350	800	-
HI-TRAC top lid neutron shielding	300	350	-
HI-TRAC radial neutron shield	307	N/A	-
HI-TRAC radial lead gamma shield	350	600	-
Remainder of HI-TRAC	350	800	-
Fuel Cladding	752	752 or 1058 (Short Term Operations)†† 1058 (Off-Normal and Accident Conditions)	752
Overpack concrete	300	572 (local temperature	450 (local temperature)
Overpack Lid Top and Bottom Plate	450	800	450
Remainder of overpack steel structure	350	800	450

† 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^{O}F. For the ISFSI fire event, the maximum temperature limit for ASME Section 1 equipment is appropriate (850^{O}F in Code Table 1A).

* 30-day accident event is defined as a 100% blocked vent condition at threshold heat loads defined in Section 4.6.

†† Normal short term operations includes MPC drying and onsite transport per Reference [2.0.8]. The 1058^{o}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^{o}F. See also Section 4.3.

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

Fuel Assembly Array and Class	**10x10F**	**10x10G**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 188
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 8, 9)	4.7 (Note 7)	4.75 (Note 7)
Initial Rod Maximum Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 4)	96/84
Fuel Clad O.D. (in.)	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150
No. of Water Rods (Note 6)	2	5 (Note 5)
Water Rod Thickness (in.)	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.120	≤ 0.060

Table 2.III.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. See Table glossary for the definition of “ZR.”

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

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

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

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

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

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

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

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

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

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

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

Since both the HI-STORM 100 and HI-TRAC may serve as an MPC carrier, their lifting attachments are designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and Regulatory Guide 3.61 [1.0.2].

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

3.1.2 Design Criteria

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

Consistent with the provisions of NUREG-1536, the central objective of the structural analysis

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

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

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

Finally, the lifting attachments or the interfacing lifting points in the HI-STORM 100 Overpack and HI-TRAC casks and the multi-purpose canisters must meet the requirements of NUREG-0612 and/or Regulatory Guide 3.61 as described in Subsection 3.4.3 and Tables 2.0.1, 2.0.2 and 2.0.3 under a normal handling condition (Load Case 01 in Table 3.1.5). The load combination D+H in Table 3.1.5 is equivalent to 1.15D. This is further explained in Subsection 3.4.3.

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

3.1.2.3 Brittle Fracture

The MPC canister and basket are constructed from a series of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum temperature range of the HI-STORM 100 System. Therefore, brittle fracture is not a concern for the MPC components. Such an assertion cannot be made a priori for the HI-STORM storage

overpack and HI-TRAC transfer cask that contain ferritic steel parts. In general, the impact testing requirements for the HI-STORM overpack and the HI-TRAC transfer cask are a function of two parameters: the Lowest Service Temperature (LST) and the normal stress level. The significance of these two parameters, as they relate to impact testing of the overpack and the transfer cask, is discussed below.

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

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

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

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

It is well known that the NDT temperature of steel is a strong function of its composition, manufacturing process (viz., fine grain vs. coarse grain practice), thickness, and heat treatment. For example, according to Burgreen [3.1.3], increasing the carbon content in carbon steels from 0.1% to 0.8% leads to the change in NDT from -50°F to approximately 120°F. Likewise, lowering of the normalizing temperature in the ferritic steels from 1200°C to 900°C lowers the NDT from 10°C to -50°C [3.1.3]. It, therefore, follows that the fracture toughness of steels can be varied significantly within the confines of the ASME Code material specification set forth in Section II of the Code. For example, SA516 Gr. 70 (which is a principal NF material in the HI-STORM 100 Overpack) can have a maximum carbon content of up to 0.3% in plates up to four inches thick. Section II further permits normalizing or quenching followed by tempering to enhance fracture toughness. Manufacturing processes which have a profound effect on fracture toughness, but little effect on tensile or yield strength of the material, are also not specified with the degree of specificity in the ASME Code to guarantee a well defined fracture toughness. In fact, the Code relies on actual coupon testing of the part to ensure the desired level of protection

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

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

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

i. Trunnion: NUREG-0612 and Reg. Guide 3.6.1 require that under the "apparent dead load", D^*, the maximum primary stress in the trunnion be less than 10% of the trunnion material ultimate strength and less than 1/3rd of the trunnion material yield strength. Because of the materials of construction selected for trunnions in all HI-STORM 100 System components, the ultimate strength-based limit is more restrictive in every case. Therefore, all trunnion safety factors reported in this document pertain to the ultimate strength-based limit.

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

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

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

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

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

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

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

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

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

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

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

3.4.3.1 125 Ton HI-TRAC Lifting Analysis - Trunnions

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

Specifically, the following results are obtained:

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

Note that the safety factor presented in the previous table represents the additional margin beyond the mandated limit of 3 on yield strength and 10 on tensile strength.

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

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

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

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

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

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

Notes:

1. Safety factor is for MPC baseplate.

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

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

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

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

S = 2 x 1,633 psi

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

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

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

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

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

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

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

The wall thickness of the storage cells, which is by far the most significant variable in a fuel basket's structural strength, is significantly greater in the MPCs than in comparable fuel baskets licensed in the past. In spite of their relatively high flexural rigidities, computed margins in the fuel baskets are rather modest. This is because of some assumptions in the analysis that lead to an overstatement of the state of stress in the fuel basket. For example:

i. The section properties of longitudinal fillet welds that attach contiguous cell walls

additional stability margin.

The penetration potential of the missile strikes (Load Case 04 in Table 3.1.5) is examined first. The detailed calculations show that there will be no penetration through the concrete surrounding the inner shell of the storage overpack or penetration of the top closure plate. Therefore, there will be no impairment to the confinement boundary due to missile strikes during a tornado. Since the inner shell is not compromised by the missile strike, there will be no permanent deformation of the inner shell. Therefore, ready retrievability is assured after the missile strike. The following paragraphs summarize the analysis work for the HI-STORM 100.

a. The small missile will dent any surface it impacts, but no significant puncture force is generated. The 1" missile can enter the air ducts, but geometry prevents a direct impact with the MPC.

b. The following table summarizes the denting and penetration analysis performed for the intermediate missile. Denting is used to connote a local deformation mode encompassing material beyond the impacting missile envelope, while penetration is used to connote a plug type failure mechanism involving only the target material immediately under the impacting missile. The results are applicable to the HI-STORM 100 and to the HI-STORM 100S. The HI-STORM 100S version B has a thicker outer shell than the classic HI-STORM 100, and a lid configuration that consists of a 1” lid cover plate backed by concrete and a 3” thick lid vent shield plate that acts as a barrier to a top lid missile strike. Therefore, the tabular results presented below are bounding for the HI-STORM 100S Version B.

Location	Denting (in.)	Thru-Thickness Penetration
Storage overpack outer Shell	13.75†	Yes (>0.75 in.)
Radial Concrete	18.54††	No (<27.25 in.)
Storage overpack Top Lid	< 2.0	No (<4 in.)

† Based on minimum outer shell thickness of 3/4”. Penetration is less for HI-STORM 100 and 100S overpacks with 1” thick outer shell.

†† Based on concrete compressive strength equal to 50% of minimum value specified in Table 3.3.5 to account for exposure to high temperatures resulting from blocked duct accident.

The primary stresses that arise due to an intermediate missile strike on the side of the storage overpack and in the center of the storage overpack top lid are determined next. The analysis of the storage lid for the HI-STORM 100 bounds that for the HI-STORM 100S; because of the

additional energy absorbing material (concrete) in the direct path of a potential missile strike on the top lid of the HI-STORM 100S lid, the energy absorbing requirements of the circular plate structure are much reduced. The analysis demonstrates that Level D stress limits are not exceeded in either the overpack outer shell or the top lid. The safety factor in the storage overpack, considered as a cantilever beam under tip load, is computed, as is the safety factor in the top lids, considered as two centrally loaded plates. The applied load, in each case, is the missile impact load. Similar calculations are performed for the HI-STORM 100S Version B using the same model and methodology. A summary of the results for axial stress in the storage overpack is given in the table below with numbers in parentheses representing the results of calculations for the geometry of the HI-STORM 100S Version B:

HI-STORM 100 MISSILE IMPACT - Global Axial Stress Results			
Item	**Value (ksi)**	**Allowable (ksi)**	**Safety Factor**
Outer Shell – Side Strike	14.35†(15.17)	37.95	2.64†(2.50)
Top Lid - End Strike	44.14(47.57)	57.0 (50.65)	1.29(1.065)

† Based on HI-STORM 100 overpack with inner and outer shell thicknesses of 1-1/4" and 3/4", respectively. Result is bounding for HI-STORM 100 overpacks made with 1" thick inner and outer shells because the section modulus of the steel structure is greater.

To demonstrate ready retrievability of the MPC, we must show that the storage overpack suffers no permanent deformation of the inner shell that would prevent removal of the MPC after the missile strike. To demonstrate ready retrievability (for both HI-STORM 100 and for HI-STORM 100S) a conservative evaluation of the circumferential stress and deformation state due to the missile strike on the outer shell is performed. A conservative estimate for the 8" diameter missile impact force, "Pi", on the side of the storage overpack is calculated as:

Pi = 843,000 lb.

This force is conservative in that the target overpack is assumed rigid; any elasticity serves to reduce the peak magnitude of the force and increase the duration of the impact. The use of the upper bound value is the primary reason for the high axial stresses resulting from this force. To demonstrate continued ability to retrieve the MPC subsequent to the strike, circumferential stress and deformation that occurs locally in the ring section near the location of the missile strike are investigated.

Subsection 3.4.7 presents stress and displacement results for a composite ring of unit width consisting of the inner and outer shells of the storage overpack. The solution assumes that the net loading is 56,184 lb. applied on the 1" wide ring (equivalent to a 45g deceleration applied uniformly along the height on a storage overpack weight of 270,000 lb.). This solution can be applied directly to evaluate the circumferential stress and deformation caused by a tornado missile strike on the outer shell. Using the results for the 45g tipover event, an attenuation factor to adjust the results is developed that reflects the difference in load magnitude and the width of

the ring that is effective in resisting the missile strike force. The strike force Pi is resisted by a combination of inertia force and shear resistance from the portion of the storage overpack above and below the location of the strike. The ring theory solution to determine the circumferential stress and deformation conservatively assumes that inertia alone, acting on an effective length of ring, balances the applied point load Pi. The effective width of ring that balances the impact load is conservatively set as the diameter of the impacting missile (8") plus the effect of the "bending boundary layer" length. This boundary layer length is conservatively set as a multiple of twice the square root of the product of mean radius times the average thickness of two shells making up the cylindrical body of the storage overpack. The mean radius of the composite cylinder and the average thickness of the inner and outer shells are

R_{mean} = 48"

T =.5 x (.75"+1.25") = 1"

The bending boundary layer "β" in a shell is generally accepted to be given as $(2(R_{mean}T)^{1/2})$ = 13.85" for this configuration. That is, the effect of a concentrated load is resisted mainly in a length along the shell equal to the bending boundary layer. For a strike away from the ends of the shell, a boundary layer length above and below the strike location would be effective (i.e., double the boundary layer length). However, to conservatively account for resistance above and below the location of the strike, this calculated result is only increased by 1.5 in the following analysis (rather than 2). Therefore, the effective width of ring is assumed as:

13.85" x 1.5 + 8" = 28.78"

The solution for the 45g tipover event (performed for a unit ring width and a load of 56,184 lb.) is directly applicable if we multiply all stress and displacement results by the factor "Y" where

Y = (1"/28.78") x (843,000 lb./56,184 lb.) = 0.521

Using this factor gives the following bounding results for maximum circumferential stresses (without regard for sign and location of the stress) and deformations due to the postulated tornado missile strike on the side of the storage overpack outer shell:

Maximum circumferential stress due to bending moment = (29,310 psi x Y) = 15,271 psi

Maximum circumferential stress due to mean tangential force = (18,900 lb./2 sq.inch) x Y = 4,923 psi

Change in diameter in the direction of the load = -0.11" x Y = -0.057"

Change in diameter perpendicular to the direction of the load = +0.06" x Y = 0.031"

Based on the above calculation, the safety factor on maximum stress for this condition is

SF = 37,950psi/15,271 psi = 2.48

The allowable stress for the above calculation is the Level D membrane stress intensity limit from Table 3.1.12 at 450°F. This is a conservative result since the stress intensity is localized and need not be compared to primary membrane stress intensity. Even with the overestimate of impact strike force used in the calculations here, the stresses remain elastic and the calculated diameter changes are small and do not prevent ready retrievability of the MPC. Note that because the stresses remain in the elastic range, there will be no post-strike permanent deformation of the inner shell.

The above calculations remain valid for the HI-STORM 100S, Version B using normal weight concrete and are bounding for the case where densified concrete is used.

3.4.8.2 HI-TRAC Transfer Cask

3.4.8.2.1 Intermediate Missile Strike

HI-TRAC is always held by the handling system while in a vertical orientation completely outside of the fuel building (see Chapter 2 and Chapter 8). Therefore, considerations of instability due to a tornado missile strike are not applicable. However, the structural implications of a missile strike require consideration.

The penetration potential of the 8" missile strike on HI-TRAC (Load Case 04 in Table 3.1.5) is examined at two locations:

1. the lead backed outer shell of HI-TRAC.
2. the flat transfer lid consisting of multiple steel plates with a layer of lead backing.

In each case, it is shown that there is no penetration consequence that would lead to a radiological release. The following paragraphs summarize the analysis results.

a. The small missile will dent any surface it impacts, but no significant puncture force is generated.

b. The following table summarizes the denting and penetration analysis performed for the intermediate missile. Denting connotes a local deformation mode encompassing material beyond the impacting missile envelope, while penetration connotes a plug type failure mechanism involving only the target material immediately under the impacting missile. Where there is through-thickness penetration, the lead and the inner plate absorb any residual energy remaining after penetration of the outer plate in the 100 Ton HI-TRAC transfer lid. The table summarizes the bounding results for both transfer casks.

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

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

Notes:

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

TABLE 3.4.9
SAFETY FACTORS FROM SUPPLEMENTARY CALCULATIONS

Item	Loading	Safety Factor	FSAR Location Where Details are Provided
HI-STORM Top Lid Weld Shear	Tipover	3.22	3.4.4.3.2.2
HI-STORM Lid Bottom Plate	End Drop	9.777	3.4.4.3.2.3
HI-STORM Lid Bottom Plate Welds	End Drop	2.695	3.4.4.3.2.3
Pedestal Shield Compression	End Drop	1.011	3.4.4.3.2.3
HI-STORM Inlet Vent Plate Bending Stress	End Drop	1.271	3.4.4.3.2.3
HI-STORM Lid Top Plate Bending	End Drop –100 100S	5.208 1.357	3.4.4.3.2.3
HI-TRAC Pocket Trunnion Weld	HI-TRAC Rotation	2.92	3.4.4.3.3.1
HI-TRAC 100 Optional Bolts - Tension	HI-TRAC Rotation	1.11	3.4.4.3.3.1
HI-STORM 100 Shell	Seismic Event	14.6	3.4.7
HI-TRAC Transfer Lid Door Lock Bolts	Side Drop	2.387	3.4.4.3.3.3
HI-TRAC Transfer Lid Separation	Side Drop	1.159	3.4.4.3.3.3
HI-STORM 100 Top Lid	Missile Impact	1.065	3.4.8.1
HI-STORM 100 Shell	Missile Impact	2.50	3.4.8.1
HI-TRAC Water Jacket –Enclosure Shell Bending	Pressure	1.85	3.4.4.3.3.4
HI-TRAC Water Jacket – Enclosure Shell Bending	Pressure plus Handling	1.80	3.4.4.3.3.1
HI-TRAC Water Jacket – Bottom Flange Bending	Pressure	1.39	3.4.4.3.3.4
HI-TRAC Water Jacket – Weld	Pressure	1.42	3.4.4.3.3.4
Fuel Basket Support Plate Bending	Side Drop	1.82	3.4.4.3.1.8
Fuel Basket Support Leg Stability	Side Drop	4.07	3.4.4.3.1.8
Fuel Basket Support Welds	Side Drop	1.35	3.4.4.3.1.8
MPC Cover Plates in MPC Lid	Normal Condition Internal Pressure	1.81	3.4.4.3.1.8
MPC Cover Plate Weld	Accident Condition Internal Pressure	2.52	3.4.4.3.1.8
HI-STORM Storage Overpack	External Pressure	2.88	3.4.4.5.2
HI-STORM Storage Overpack Circumferential Stress	Missile Strike	2.60	3.4.8.1
HI-TRAC Transfer Cask Circumferential Stress	Missile Strike	2.61	3.4.8.2
HI-TRAC Transfer Cask Axial Membrane Stress	Side Drop	1.52	3.4.9.3

3.8 REFERENCES

[3.1.1] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," United States Nuclear Regulatory Commission.

[3.1.2] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, Inc.

[3.1.3] D. Burgreen, "Design Methods for Power Plant Structures", Arcturus Publishers, 1975.

[3.1.4] Deleted.

[3.1.5] Deleted.

[3.1.6] Aerospace Structural Metals Handbook, Manson.

[3.3.1] ASME Boiler & Pressure Vessel Code, Section II, Part D, 1995.

[3.3.2] American Concrete Institute, "Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)".

[3.3.3] American Concrete Institute, "Code Requirements for Nuclear Safety Related Structures" (ACI-349-85) and Commentary (ACI-349R-85)(For anchored casks, the requirements on the design of the steel embedment are ACI-349-97, including Appendix B and the Commentary (ACI-349R-97)).

[3.3.5] J.H. Evans, "Structural Analysis of Shipping Casks, Volume 8, Experimental Study of Stress-Strain Properties of Lead Under Specified Impact Conditions", ORNL/TM-1312, Vol. 8, ORNL, Oak Ridge, TN, August, 1970.

[3.4.1] ANSYS 5.3, ANSYS, Inc., 1996 (Current usage of ANSYS includes Versions up thru 7.0, 2003).

[3.4.2] ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, 1995.

[3.4.3] ASME Boiler & Pressure Vessel Code, Section III, Appendices, 1995.

[3.4.4] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, 1995.

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

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

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

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

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

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

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

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

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

3.I.4.4 Heat

i. Summary of Pressures and Temperatures

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

ii. Differential Thermal Expansion

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

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

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

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

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

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

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

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

Finally, to evaluate the potential for crack propagation and growth for the MPC-68M fuel basket under the non-mechanistic tipover event, a bounding crack propagation analysis is carried out in Attachment D of [1.III.A.3]. The analysis demonstrates that a through-thickness linear flaw measuring 1/32 inches in length (i.e., maximum undetectable flaw size per inspection criteria) remains stable under the most severe accident loading conditions.

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

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

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

3.III.8 REFERENCES

[3.III.1] Buckling of Bars, Plates, and Shells, D.O. Brush and B.O. Almroth, McGraw-Hill, 1975, p.22.

[3.III.2] Properties of Aluminum Alloys, Tensile, Creep, and Fatigue Data at High and Low Temperatures, ASM International, November 2006.

[3.III.3] ASME Boiler & Pressure Vessel Code, Section II, Parts A and D, American Society of Mechanical Engineers, 2007.

[3.III.4] Deleted.

[3.III.5] ASTM Specification B221M-07, "Standard Specification for Aluminum and Aluminum-Alloy Extruded Bars, Rods, Wire, Profiles, and Tubes (Metric)".

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

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

Notes:

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

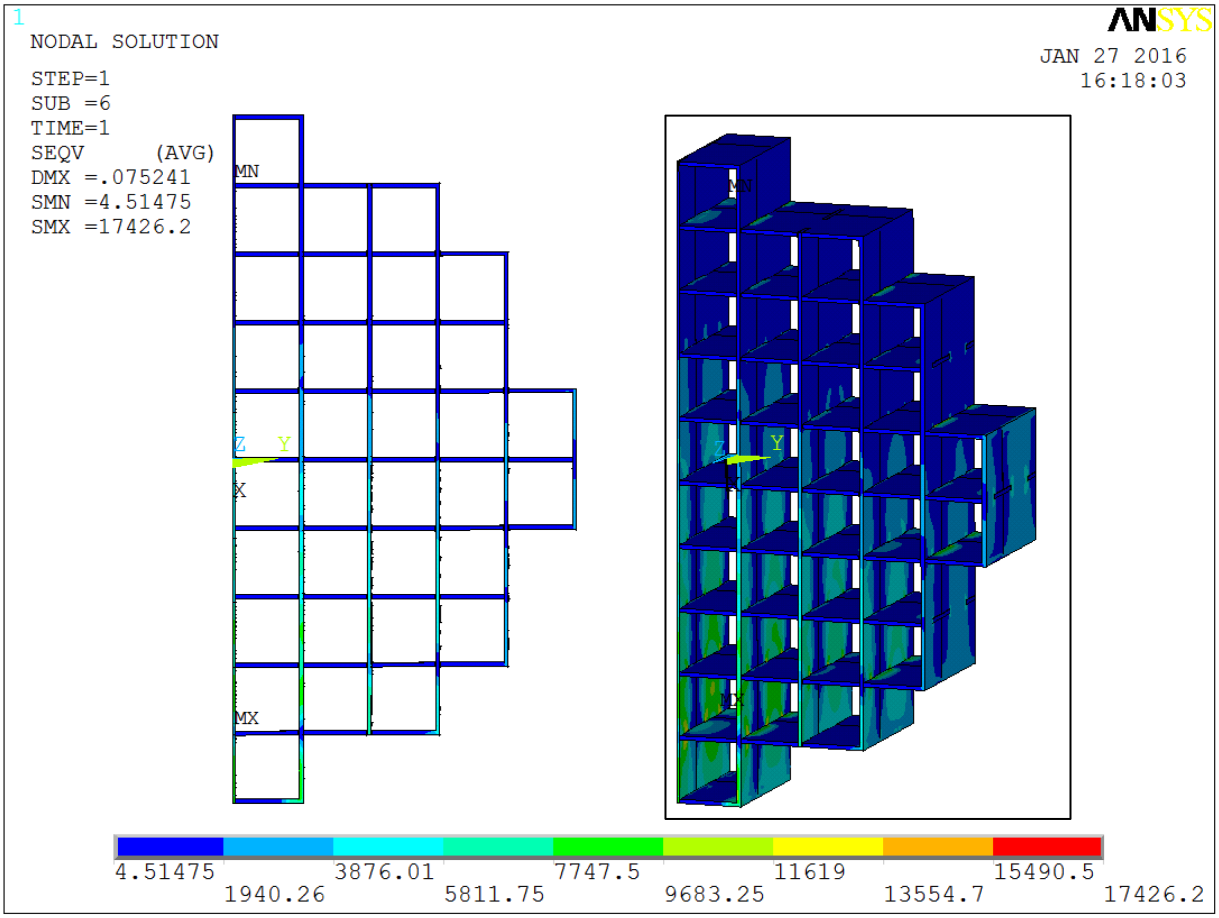

FIGURE 3.III.2: VON MISES STRESS DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60g LOAD

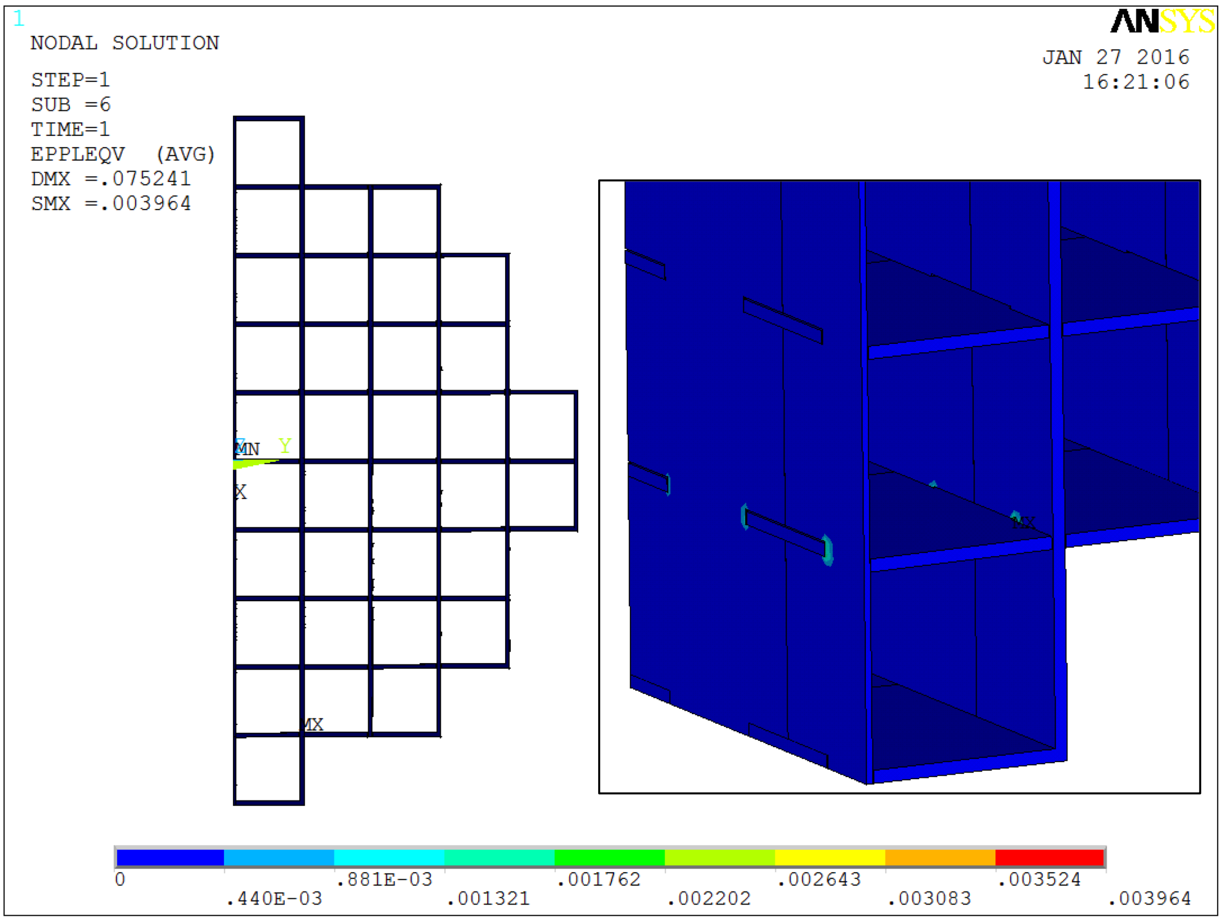

FIGURE 3.III.3: PLASTIC STRAIN DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60g LOAD

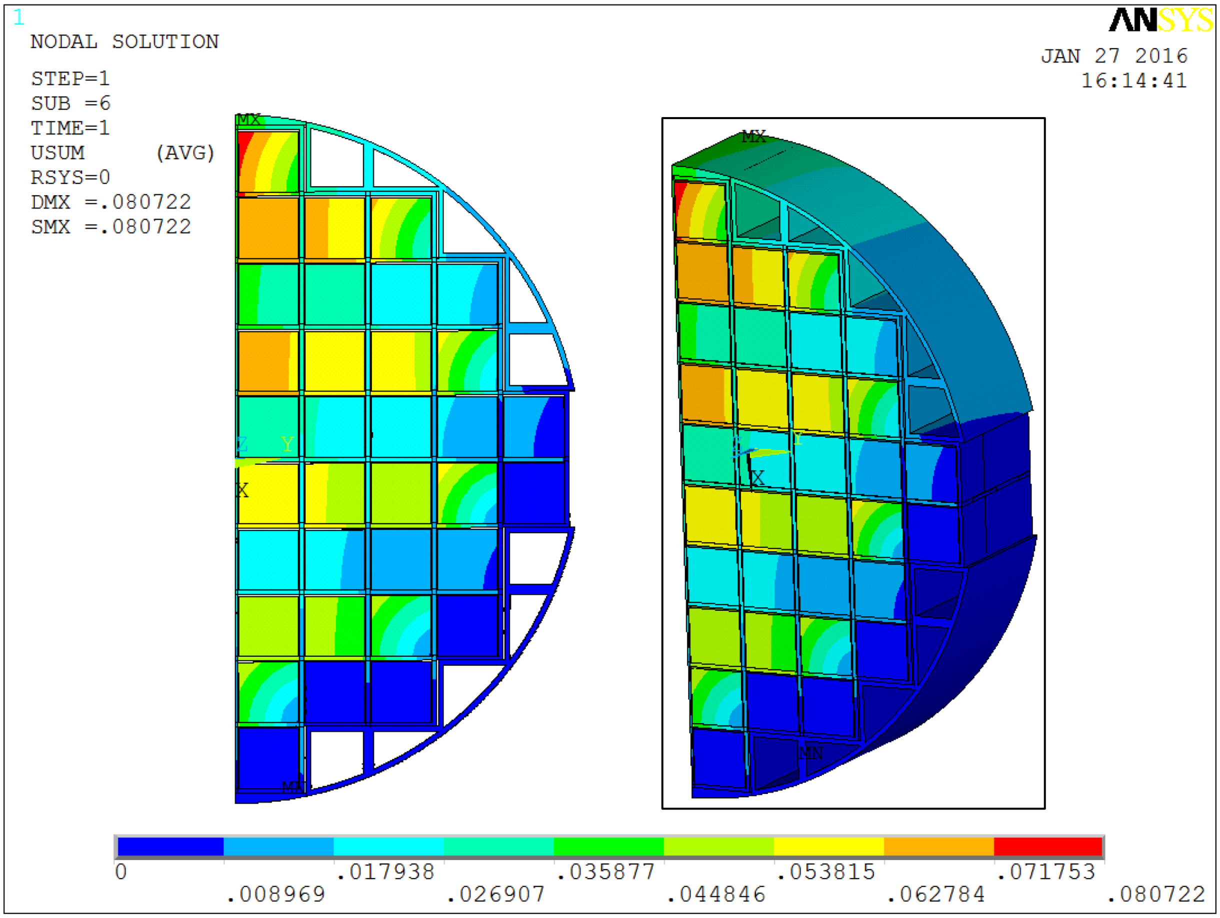

FIGURE 3.III.4: DISPLACEMENT CONTOURS IN MPC-68M FUEL BASKET UNDER 60g LOAD

Elevation (ft)	Pressure (psia)	Ambient Temperature Reduction versus Sea Level
Sea Level (0)	14.70	0°F
2000	13.66	7.1°F
4000	12.69	14.3°F

A survey of the elevation of nuclear plants in the U.S. shows that nuclear plants are situated near about sea level or elevated slightly (~1000 ft). The effect of the elevation on peak fuel cladding temperatures is evaluated by performing calculations for a HI-STORM 100 System situated at an elevation of 1500 feet. At this elevation the ambient temperature would decrease by approximately 5°F (See Table above). The peak cladding temperatures are calculated for a bounding configuration (non-uniform storage at X = 0.5), and conservatively assuming no reduction in ambient temperature using the 3D model described in Subsection 4.4.1.1 and compared to the sea level conditions. The results are given in the following table.

MPC Design	PCT at Sea Level	PCT at 1500 feet
MPC-32 PWR	711.4°F	723.8°F
MPC-68 BWR	697.1°F	718.2°F

These results show that the PCT, including the effects of site elevation, continues to be well below the regulatory cladding temperature limit of 752°F. In light of the above evaluation, it is not necessary to place any ISFSI elevation constraints for HI-STORM deployment at elevations up to 1500 feet. If, however, an ISFSI is sited at an elevation greater than 1500 feet, the effect of altitude on the PCT shall be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient conditions.

4.4.5 Maximum Internal Pressure

4.4.5.1 MPC Helium Backfill Pressure

For design basis heat load, the helium backfill shall be sufficient to produce the required operating pressure of 7 atmospheres (absolute) during normal storage at reference conditions (See Table 4.0.1). Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain helium backfill pressure specified at a reference temperature (70°F). The minimum backfill pressure to attain this operating pressure for each MPC type is provided in Table 4.4.11. An upper limit on the helium backfill pressure corresponds to the design pressure of the MPC vessel (Table 2.2.1). The upper limit on the backfill pressure is also reported in Table 4.4.11. To bound the minimum and maximum backfill pressures listed in Table 4.4.11 with margin, a helium backfill specification is set forth in Table 4.4.12. These values support the technical specification of the system for the design basis heat load of the MPC.

In addition the technical specifications allow for using a wider range on the backfill pressure if the heat load of the MPC is less than 28.74 kW. The minimum of this range corresponds to an operating

Table 4.5.10

PRINCIPAL SITE-SPECIFIC TIME-TO-BOIL MODELING STEPS

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

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

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

30-Day 100% Vent Blockage Accident

As noted above, the fuel and component temperatures rise due to complete blockage of HI-STORM vents. This temperature rise is small for casks where heat loads are much lower than design basis heat loads. A threshold heat load is defined for all MPCs in Table 4.6.8 at or below which fuel and component temperatures remain below their respective 30-day accident temperature limits (Table 2.2.3) under steady state conditions. A steady state evaluation of a complete vent blockage at threshold heat loads is performed for both MPC-32 and MPC-68. Steady state temperature and MPC cavity pressure results are presented in Table 4.6.9. The results demonstrate that the fuel and component temperatures remain below their respective 30-day accident temperature limits defined in the Design Criteria Chapter 2 with robust margins. MPC cavity pressure is also below the accident design limit with robust margins. Thermal performance of MPC-68 bounds all types of MPC-68 and MPC-24. Therefore, the threshold total decay heat for MPC-68 is also adopted for all other variants of MPC-68 and MPC-24 canisters. To identify and clear any blockages mandatory surveillance is defined in Chapter 11.

Since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, the following evaluations are performed to demonstrate that the MPCs are safe at off-normal and accident conditions. Thermal off-normal and design basis events or accident conditions defined in Chapter 4 of the FSAR are concurrently evaluated with the 100% vent blockage event at threshold heat load:

(a) Pressure (fuel rod rupture): There is no credible event to cause fuel rods to rupture during a 100% vent blockage event because of the following reasons:

- The computed PCT under 100% vent blockage accident condition (Table 4.6.9) is below the ISG-11 Rev 3 long-term normal temperature limit, and
- there is no credible loading on the fuel assemblies to cause fuel rods to rupture during a 100% vent blockage event.

Accordingly, the computed cavity pressures under 30-day vent blockage event evaluated herein is not affected.

(b) Off-Normal Ambient Temperatures: This event is defined in Section 4.6.1.2 as an ambient temperature of 100^{o}F for a 3-day period. The results of off-normal environmental temperatures

coincident with 100% vent blockage event are summarized in Table 4.6.12. Component temperatures are obtained by adding the off-normal-to-normal ambient temperature difference of 20^{o}F (11.1^{o}C) to temperatures computed for MPCs at threshold decay heat (Table 4.6.9). The results are below the off-normal limits (Table 2.2.3) with substantial margins.

(c) Partial Blockage of Air Inlets: This condition is already covered by the postulated event wherein all the HI-STORM vents are assumed blocked.

(d) Fire: During transfer operations at the ISFSI, there is a possibility of a fire accident event to occur coincident with a 100% vent blockage event. The impact of fire on the MPC and fuel temperatures is extremely small (approximately 1^{o}F). Overpack temperatures are primarily impacted due to heat input from the fire which is considerably larger than the SNF decay heat. Therefore, as evaluated in Section 4.6.2.1(a), the overpack components and concrete temperatures remain below their respective accident temperature limits. Therefore, this accident event coincident with a 100% vent blockage event does not challenge the HI-STORM 100 System safety limits.

(e) Extreme Environment Temperature: This event is defined in Section 4.6.2.3 as an ambient temperature of 125^{o}F for a 3-day period. The results of extreme environmental temperatures coincident with 100% vent blockage event are summarized in Table 4.6.13. Component temperatures are obtained by adding the extreme-to-normal ambient temperature difference of 45^{o}F (25^{o}C) to temperatures computed for MPCs at threshold decay heat (Table 4.6.9). The results are below the accident limits (Table 2.2.3) with substantial margins.

(f) Burial under Debris: This accident event is evaluated in Section 4.6.2.5. Since the threshold decay heat is substantially lower than the maximum design basis heat load and cask initial temperatures (Table 4.6.9) are similar for 100% vent blockage event and that evaluated in Table 4.6.6, the evaluation in Section 4.6.2.5 remains bounding.

In this manner the above evaluations reasonably assure that the HI-STORM 100 system containing MPCs are safe under off-normal and accident conditions coincident with 30-day 100% blocked vents under the threshold heat load.

4.6.2.5 Burial Under Debris

Burial of the HI-STORM 100 System under debris is not a credible accident. During storage at the ISFSI there are no structures over the casks. Minimum regulatory distances from the ISFSI to the nearest ISFSI security fence precludes the close proximity of substantial amount of vegetation. There is no credible mechanism for the HI-STORM 100 System to become completely buried under debris. However, for conservatism, complete burial under debris is considered.

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 System will undergo a transient heat up under adiabatic conditions. The

Table 4.6.8
THRESHOLD DECAY HEAT FOR 100% VENT BLOCKAGE

MPC Type	Threshold Decay Heat, kW	Per Storage Cell Decay Hear Limit, kW
MPC-24/24E/EF	18	0.75
MPC-68/68F/68FF/68M	18	0.264
MPC-32/32F	16	0.5

Table 4.6.9
STEADY STATE MAXIMUM HI-STORM TEMPERATURES AND MPC CAVITY PRESSURE AT THRESHOLD HEAT LOAD UNDER 100% VENT BLOCKAGE

Component	**MPC-32 Temperatures (°F)**	**MPC-68 Temperatures (°F)**
Fuel Cladding	714	730
MPC Basket	712	727
MPC Shell	471	502
MPC Lid (Note 1)	495	522
MPC Closure Ring	453	486
MPC Baseplate (Note 1)	327	342
Overpack Inner Shell (Note 2)	403	430
Overpack Concrete	401	426
Overpack Lid Concrete Bottom Plate	372	396
Overpack Lid Concrete Top Plate	221	225
Overpack Lid Concrete	372	396
MPC Cavity Pressure, psig (Note 3)		
No Rod Rupture With 1% Rod Rupture	102.6 103.6	104.7 105.2

Note 1: Thru-thickness section average temperature is reported.
Note 2: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.
Note 3: Although the CFD evaluations have been performed with an operating temperature corresponding to minimum initial helium backfill specification of 29.3 psig at 70°F reference temperature, maximum initial helium backfill pressure of 48.5 psig is adopted to compute MPC cavity pressure. In reality, the actual MPC cavity pressure will be lower than that reported above.

Table 4.6.10

PRINCIPAL SITE-SPECIFIC HI-STORM FIRE ACCIDENT MODELING STEPS

Step 1: Site Specific Conditions	Heat Loads Site Specific heat load map. Ambient Temperature – Normal storage temperature defined in Chapter 2. Fire Accident – Compute fire duration τ_f based on site specific fuel quantity in accordance with methodology defined in Sub-Section 4.6.2.1(a).
Step 2: FLUENT Thermal Model	Incorporate HI-STORM thermal methodologies defined in Sub-Sections 4.4.1.1 and 4.4.1.2. Use the licensing basis HI-STORM thermal model presented in [4.5.1]. Apply heat loads and ambient temperature defined in Step 1 and obtain baseline initial temperature field.
Step 3: Fire Transient Solution	Apply fire parameters defined by fire temperature, fire emissivity and convection heat transfer coefficient specified in Sub-Section 4.6.2.1(a) to FLUENT Model and compute time dependent HI-STORM temperature field starting from initial temperature field obtained in Step 2 upto end of fire τ_f.
Step 4: Post-Fire Solution	Restore ambient temperature conditions as defined in Sub-Section 4.6.2.1(a) and compute time dependent temperature field under cooldown of HI-STORM cask by natural convection and radiation. Conservatively assume paint loss from all exterior surfaces. Continue solution until all component and fuel temperatures reach their maximum and begin to recede.
Step 5: Post-Process Results	Post-process FLUENT solution and evaluate compliance of maximum fuel, basket, MPC confinement boundary, HI-STORM concrete and enclosure shell temperatures to Chapter 2 accident temperature limits. Additionally, steel structure of the overpack shall remain physically stable, i.e., the maximum temperature shall be less than 50% of the component's melting temperature. Compute maximum MPC pressure in accordance with Sub-Section 4.4.5 methodology and evaluate compliance with Chapter 2 accident pressure limits.

Table 4.6.11

PRINCIPAL SITE-SPECIFIC HI-TRAC FIRE ACCIDENT MODELING STEPS

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

Table 4.6.12

STEADY STATE HI-STORM TEMPERATURES WITH MPCs AT THRESHOLD HEAT LOAD UNDER 100% VENT BLOCKAGE AND COINCIDENT OFF-NORMAL ENVIRONMENTAL TEMPERATURE

Component	**MPC-32 Temperatures** Note 1 **(^oF)**	**MPC-68 Temperatures** Note 1 **(^oF)**
Fuel Cladding	734	750
MPC Basket	732	747
MPC Shell	491	522
MPC Lid (Note 2)	515	542
MPC Closure Ring	473	506
MPC Baseplate (Note 2)	347	362
Overpack Inner Shell (Note 3)	423	450
Overpack Body Concrete	421	446
Overpack Lid Bottom Plate	392	416
Overpack Lid Top Plate	241	245
Overpack Lid Concrete	392	416
MPC Cavity Pressure, psig		
With 1% Rod Rupture	106.0	107.6
Note 1: Unless otherwise specified, all the reported temperatures are peak maximum values. Note 2: Maximum through thickness average temperature at the hottest location is reported for structural thick components. Note 3: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.		

Table 4.6.13

STEADY STATE HI-STORM TEMPERATURES WITH MPCs AT THRESHOLD HEAT LOAD UNDER 100% VENT BLOCKAGE AND COINCIDENT EXTREME ENVIRONMENTAL TEMPERATURE

Component	**MPC-32 Temperatures Note 1 (^{o}F)**	**MPC-68 Temperatures Note 1 (^{o}F)**
Fuel Cladding	759	775
MPC Basket	757	772
MPC Shell	516	547
MPC Lid (Note 2)	540	567
MPC Closure Ring	498	531
MPC Baseplate (Note 2)	372	387
Overpack Inner Shell (Note 3)	448	475
Overpack Body Concrete	446	471
Overpack Lid Bottom Plate	417	441
Overpack Lid Top Plate	266	270
Overpack Lid Concrete	417	441
MPC Cavity Pressure, psig		
With 1% Rod Rupture	109.0	110.6

Note 1: Unless otherwise specified, all the reported temperatures are peak maximum values.
Note 2: Maximum through thickness average temperature at the hottest location is reported for structural thick components.
Note 3: The overpack inner shell maximum temperature bounds the temperature of the remaining overpack steel structure.

[4.5.1] "HI-STORM THERMAL-HYDRAULIC ANALYSES SUPPORTING UP TO 36.9 KW HIGH HEAT LOAD AMENDMENT", Holtec Report HI-2043317, Latest Revsion.

[4.5.2] HI-STORM FW FSAR, Holtec Report HI-2084239, Rev. 1, Section 3.4.4.1.11, Docket No. 72-1032.

[4.6.1] United States Code of Federal Regulations, Title 10, Part 71.

[4.6.2] Gregory, J.J. et. al., "Thermal Measurements in a Series of Large Pool Fires", SAND85-1096, Sandia National Laboratories, (August 1987).

CHAPTER 5†: SHIELDING EVALUATION

5.0 INTRODUCTION

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

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

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

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

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

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

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

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

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

The design basis accidents analyzed in Chapter 11 have one bounding consequence that may affect the shielding materials of the HI-STORM overpack. It is the 100% blockage of air inlets 30 day accident condition. To analyze this accident, it is conservatively assumed that all hydrogen and portion of oxygen are lost in a bounding volume of the neutron shield layer (concrete) that reaches a temperature of at least 350°F . The 100% blockage of air inlets 30 day accident affects the dose at 100 m from overpack. To illustrate the impact of this design basis accident, the dose rate at 91.44 m is provided in Table 5.1.17. The burnup and cooling time combination used in Table 5.1.17 correlates to the HI-TRAC accident condition discussed in this section and in Table 5.1.10. The heat load for this burnup and cooling time combination is significantly larger than the heat load threshold for the 30-day surveillance of the casks discussed in Section 4.6. Shielding analysis demonstrates that the HI-STORM 100% blockage of air inlets 30 day accident condition is bounded by the HI-TRAC loss of water in water jacket accident condition and therefore is in compliance with the 10CFR72.106 limits.

Other design basis accidents which may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10 for the 100-ton and 125-ton HI-TRACs at a distance of 1 meter and for the 100-ton HI-TRAC at a distance of 100 meters. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that

Table 5.1.17

DOSE RATES FROM HI-STORM FOR ACCIDENT CONDITIONS MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH BPRAs AT BOUNDING BURNUP AND COOLING TIME

91.44 METERS FROM HI-STORM		
75,000 MWD/MTU, 5.0% ENRICHMENT, AND 5-YEAR COOLING		
Top Dose Rate (mrem/hr)	**Side Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**
0.0020	0.0872	0.0891

B&W15 is an additional 17% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

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

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

5.2.5.2 BWR Design Basis Assembly

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO_2 mass. For a given array type of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table. The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.26 were analyzed at the

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

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

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

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

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

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

terms and dose rates will be conservative.

5.2.5.3 Decay Heat Loads and Allowable Burnup and Cooling Times

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

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

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

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

Different array classes or combinations of classes were analyzed separately to determine the allowable burnup as a function of cooling time for the specified allowable decay heat limits. Calculating allowable burnups for individual array classes is appropriate because even two assemblies with the same MTU may have a different allowable burnup for the same allowable cooling time and permissible decay heat. The heavy metal mass specified in Table 5.2.25 and

5.2.26 and Section 2.1.9 for the various array classes is the value that was used in the determination of the coefficients as a function of cooling time and is the maximum for the respective assembly class. Equation coefficients for each array class listed in Tables 5.2.25 and 5.2.26 were developed. In the end, the equation for the 17x17B and 17x17C array classes resulted in almost identical burnups. Therefore, in Section 2.1.9 these array classes were combined and the coefficients for the 17x17C array class were used since these coefficients produce slightly lower allowable burnups.

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

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

As mentioned above, the fuel assembly burnup and cooling times in Section 2.1.9 were calculated using the decay heat limits which are also stipulated in Section 2.1.9. The burnup and cooling times for the non-fuel hardware, in Section 2.1.9, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup, decay heat, and enrichment equations were derived without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in Section 2.1.9 regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.

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

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

Table 5.2.36

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

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

† Derived from parameters in this table.

Table 5.2.37

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

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

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

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

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

Table 5.2.38

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

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

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

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

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

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM 100 System, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accidents that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC and the 30 day 100% blockage of air inlets for the HI-STORM overpack. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. The MCNP model of the accident condition HI-STORM assumes a bounding volume of the neutron shield (concrete) loses all hydrogen and partial oxygen, and has corresponding lower density. This bounding volume correlates to all concrete in the cask body and lid that is at a temperature of at least 350°F.

5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM 100 System, including the HI-TRAC transfer casks. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 through 5.3.6 show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. Figures 5.3.1 through 5.3.3 were created with the MCNP two-dimensional plotter and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figure 5.3.7 shows a cross sectional view of the 100-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. The dimensions for the HI-STORM 100S and HI-STORM 100S Version B overpacks are also shown on Figure 5.3.10. This figure notes two different dimensions for the inner and outer shells. These values apply only to the HI-STORM 100 and 100S. In these overpacks, the inner and outer shells can be manufactured from 1.25 and 0.75 inch thick steel, respectively, or both shells can be manufactured from 1 inch thick steel. The HI-STORM 100 and 100S were modeled as 1.25 and 0.75 inch thick shells.

utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorbing materials: Boral or Metamic. Both materials are made of aluminum and B_4C powder. The Boral contains an aluminum and B_4C powder mixture sandwiched between two aluminum plates while the Metamic is a single plate. The minimum ^{10}B areal density is the same for Boral and Metamic while the thicknesses are essentially the same. Therefore, the mass of Aluminum and B_4C are essentially equivalent and there is no distinction between the two materials from a shielding perspective. As a result, Table 5.3.2 identifies the composition for Boral and no explicit calculations were performed with Metamic.

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Section 4.4 indicates that there are localized areas in the concrete in the lid of the overpack which approach 390°F. An increase in temperature from 300°F to 390°F results in an approximate 0.666% overall density reduction due to the loss of chemically unbound water. This density reduction results in a reduction in the mass fraction of hydrogen from 0.6% to 0.529% in the area affected by the temperature excursion. This is a localized effect with the maximum loss occurring at the bottom center of the lid where the temperature is the hottest and reduced loss occurring as the temperature decreases to 300°F.

Based on these considerations, the presence of localized temperatures up to 390°F in the lid concrete has a negligible effect on the shielding effectiveness of the HI-STORM 100 overpack lid.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there are two accident conditions that have any significant impact on the shielding configuration. One of these accidents is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void. The other of these accidents is the 30 day 100% blockage of air inlets. The change is that a bounding volume of the neutron shield (concrete) is conservatively assumed to lose all hydrogen and partial oxygen, and has a lower corresponding density.

estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher for the 100-ton HI-TRAC and only ~20% lower for the HI-STORM overpack than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.
Per Supplement 5.III, the effect of the design differences between the MPC-68 and MPC-68M on the dose rates is small and all results and conclusions from the MPC-68 are applicable to the MPC-68M. Therefore, the thoria rod canister is also acceptable for storage in the MPC-68M.

5.4.9 Regionalized Loading Dose Rate Evaluation

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

5.4.10 Fuel Assemblies with Stainless Steel Replacement Rods Dose Rate Evaluation

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

replacement rods in the design basis BWR assembly are approximately 33% higher at the sides and top of the HI-STORM containing the MPC-68. Therefore, fuel assemblies containing irradiated stainless steel replacement rods are acceptable for storage and, if present in a fuel assembly, need to be considered in the site specific dose calculations.

APPENDIX 5.F

Additional Information on the Burnup Versus Decay Heat and Enrichment Equation

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

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

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

Table 5.F.1

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

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

Table 5.F.2

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

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

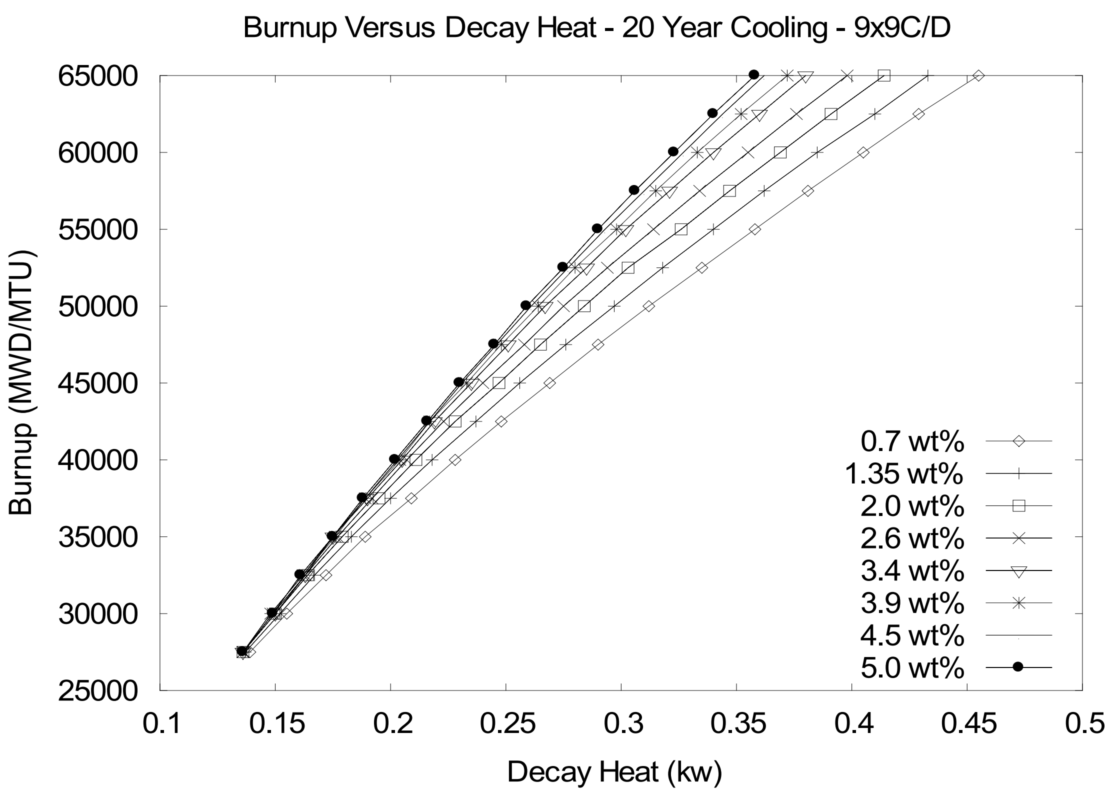

FIGURE 5.F.1; ORIGEN-S CALCULATED BURNUP VERSUS DECAY HEAT FOR VARIOUS ENRICHMENTS

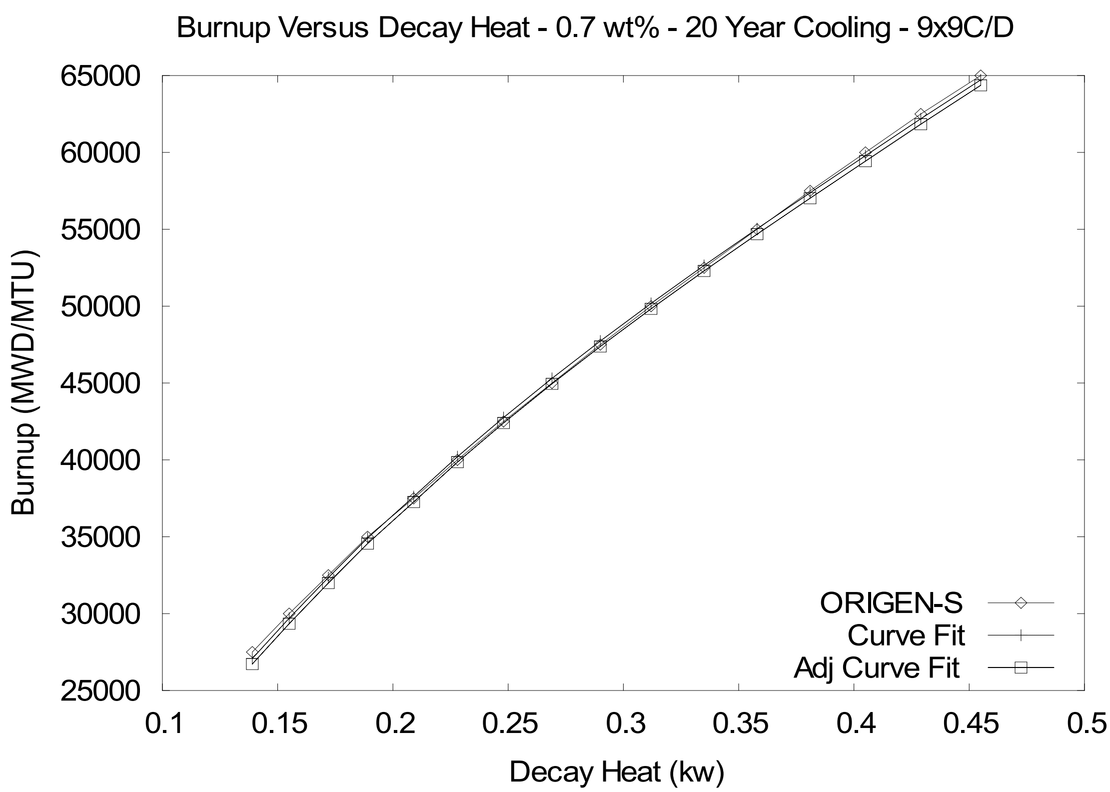

FIGURE 5.F.2; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 0.7 WT.% ^{235}U.

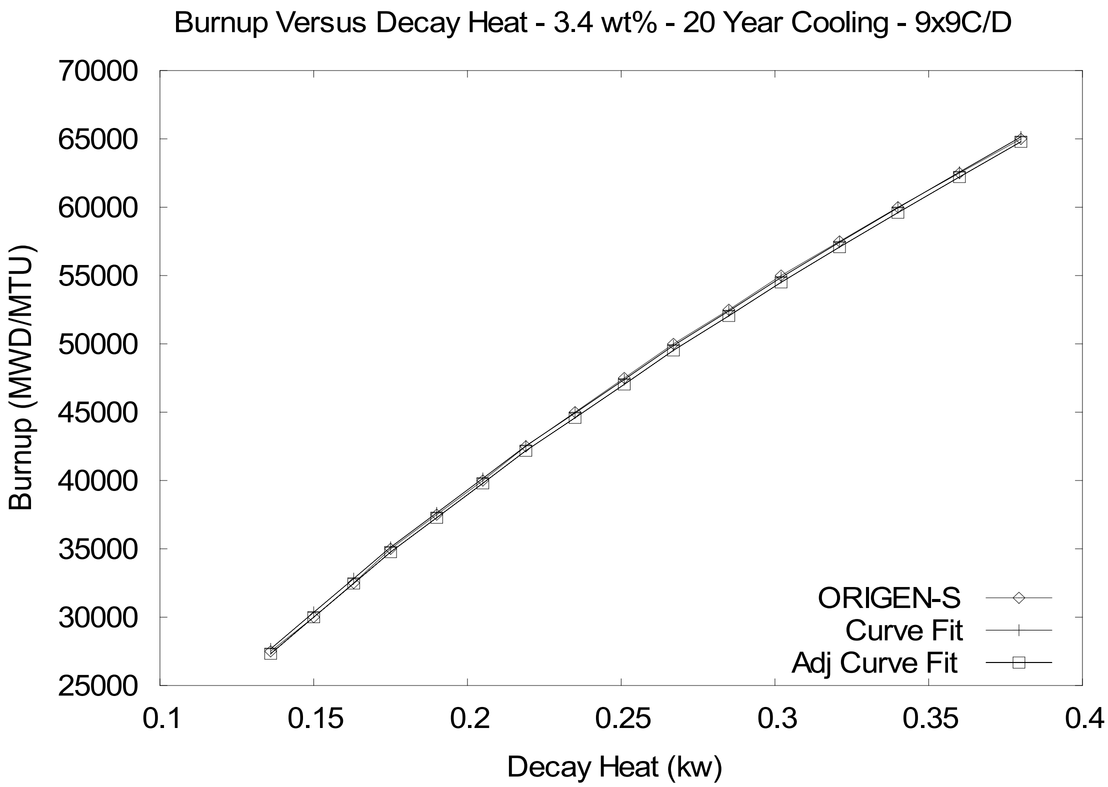

FIGURE 5.F.3; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 3.4 WT.% ^{235}U.

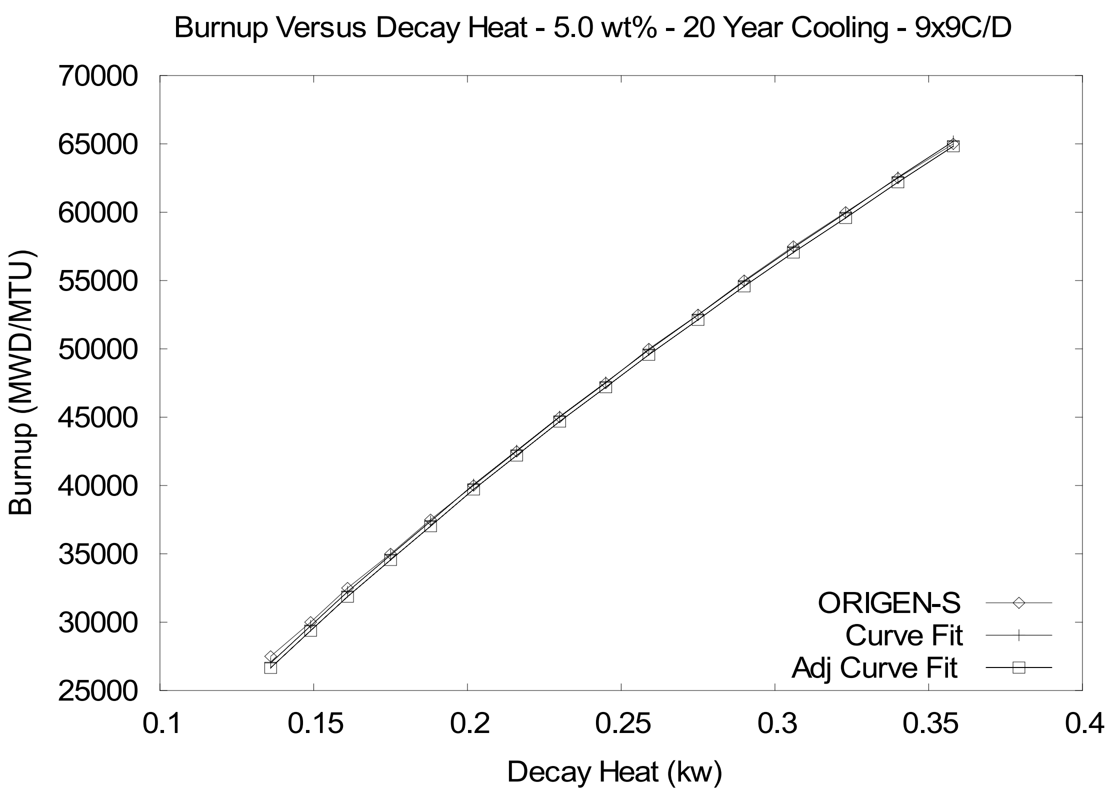

FIGURE 5.F.4; A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 5.0 WT.% ^{235}U.

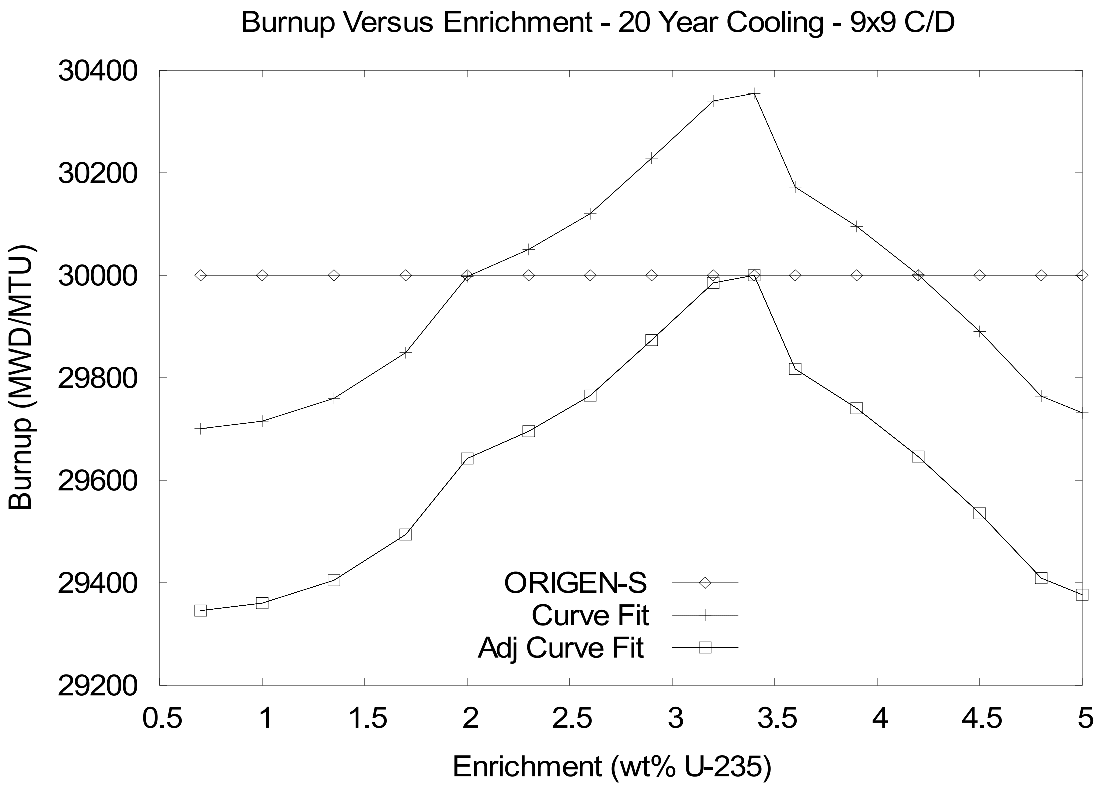

FIGURE 5.F.5; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 30,000 MWD/MTU.

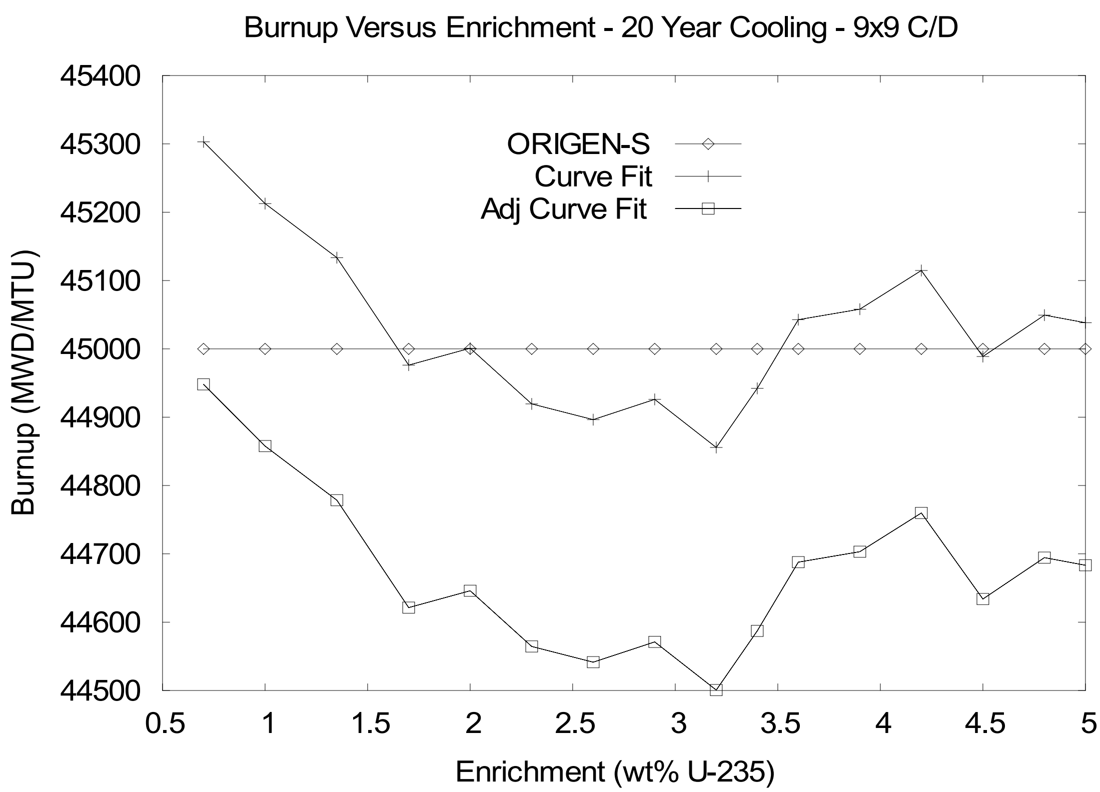

FIGURE 5.F.6; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 45,000 MWD/MTU.

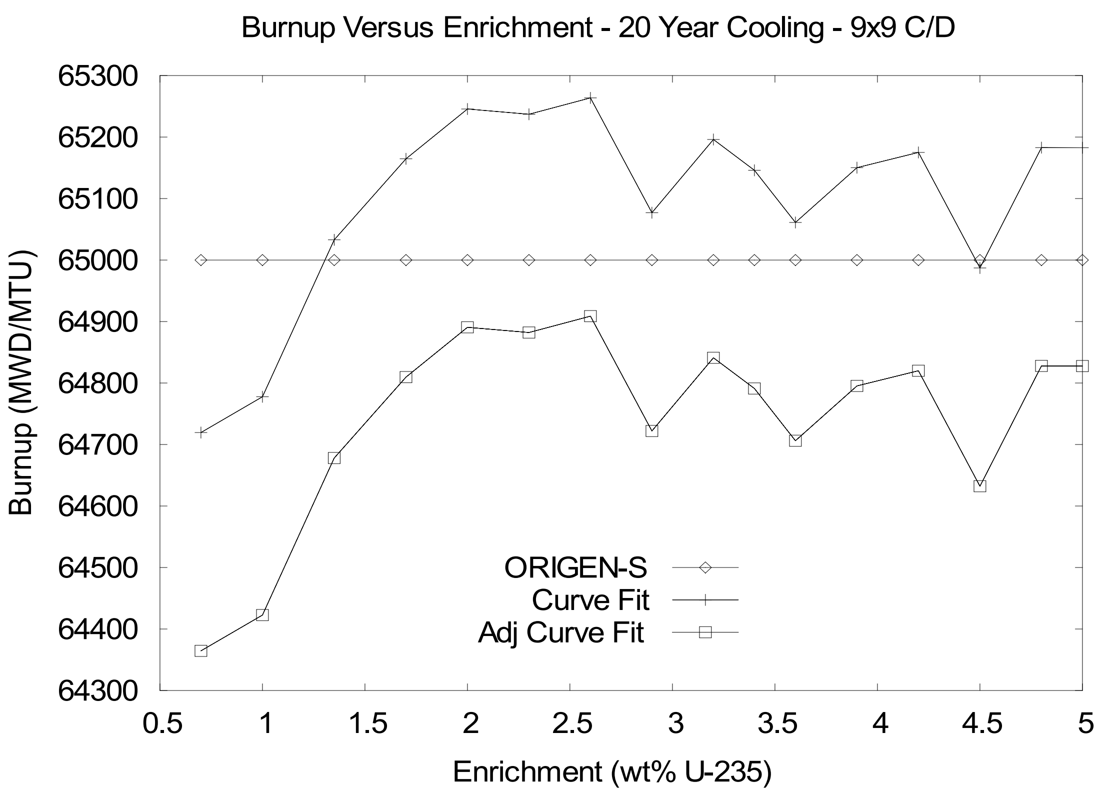

FIGURE 5.F.7; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 65,000 MWD/MTU.

STORM 100 are performed for all three combinations, and the results corresponding to the highest total dose rate at each dose location are reported. Dose rates at some locations are more dominated by the contribution from the neutron source. In this case, the highest burnup will result in the highest dose rates. At other locations, dose rates are more dominated by the contribution from the photon source terms. In this case, the shortest cooling time will result in the highest dose rates. In the result table, the burnup and cooling time combination that leads to the highest dose rate is indicated for each dose location. It should further be noted that while the number and location of assemblies with such burnup and cooling time combinations would be restricted for any actually loaded cask, it was assumed in the shielding evaluations presented here that all assemblies have the same characteristics. To indicate the level of conservatism in this approach, note that the burnup and cooling time combinations evaluated here correspond to a heat load of about 2.2 kW per assembly, or about 70 kW for the entire MPC-32. This is an important fact in respect to the choice to perform analyses for the MPC-32, and not for the other MPCs. While calculated dose rates for the other baskets could potentially be slightly higher than those for the MPC-32, based on the differences in the MPC design, fuel type, and burnup and cooling times, it is not expected that actual dose rates for any MPC would exceed those calculated here for the MPC-32. This justifies restricting the analyses here to the MPC-32. Table 5.I.1 presents the results for those burnup and cooling time combinations out of those listed above that resulted in the maximum dose rates at each location. Figure 5.I.1 identifies the locations of the dose points referenced in the table. Dose Points #1 and #2 are the locations of the inlet and outlet vents, respectively. The dose values reported adjacent to these dose points were averaged over the vent opening while the dose values reported at 1 meter from these dose locations were taken at the mid-plane of the vent. Dose Point #3, which is positioned approximately over the air flow annulus, is the location of the highest dose rate on the lid in the final storage configuration. Dose Point # 4 is averaged over the vertical air flow passage shown in Figure 5.I.1. Dose Point #4, adjacent to the overpack, is not accessible in the final storage configuration as depicted in Figure 5.I.1. Dose Point #5 is located over a tube that would be required for the ICCPS test station. Dose Point #6 is located over an empty VVM located adjacent to a loaded VVM. Except for conditions during construction discussed further below, calculations were only performed for normal conditions, since Subsections 5.1.1 and 5.1.2 concluded that, aside from the 30 day 100% blockage of air inlets accident condition, off-normal and accident conditions for the HI-STORM overpack are identical or equivalent to normal conditions for the purpose of the shielding evaluation.

The tube for the ICCPS test station is modeled as a cylindrical hole that extends from the VIP down to the base plate of the MPC. The tube is modeled with a diameter of 4 inches, located about 5.5 feet from the center of the VVM. If the actual tube has characteristics that could result in higher dose rates, i.e. is larger or closer to the VVM than modeled here, the actual tube characteristics should be considered in the site specific dose calculations. Depending on the results of those calculations, additional measures, such as added shielding at the top of the tube, may be required.

A comparison between the dose rates in Table 5.I.1 and dose rates presented in Tables 5.1.11 and 5.1.14 of the main body of this chapter show that the maximum dose rate for the HI-STORM 100U module with an MPC-32 is well below the maximum dose rate for the HI-STORM 100S

conservatively truncated to 1.0000, consistent with NUREG-1536.

- The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- Regarding the position of assemblies in the basket, configurations with centered and eccentric positioning of assemblies in the fuel storage locations are considered. For further discussions see Section 6.3.3.

- For intact fuel assemblies, as defined in Table 1.0.1, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods. The number of dummy rods used to replace missing fuel rods is not limited.

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

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

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

Table 6.1.5

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

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

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

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

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

Table 6.1.6
BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 FOR 5.0% ENRICHMENT

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

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

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

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

Table 6.1.12

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

Fuel Assembly Class of Intact Fuel	Maximum Allowable Enrichment for Intact Fuel and Damaged Fuel/Fuel Debris (wt% ^{235}U)	Minimum Soluble Boron Content (ppm) †	Maximum k_{eff}	
			HI-TRAC	HI-STAR
14x14A, B, C, D, E	4.1	1500	---	0.9336
	5.0	2300	---	0.9269
15x15A, B, C, G, I	4.1	1900	0.9349	0.9350
	5.0	2700	---	0.9365
15x15D, E, F, H	4.1	2100	---	0.9340
	5.0	2900	0.9382	0.9397
16x16A, B, C	4.1	1500	---	0.9348
	5.0	2300	---	0.9299
17x17A	4.1	1800	---	0.9311
	5.0	2600	---	0.9298
17x17B, C	4.1	2100	---	0.9294
	5.0	2900	---	0.9367

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

6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris

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

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

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

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

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

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

6.2.5 Thoria Rod Canister

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

Table 6.2.46

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

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

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

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-24 (17x17A @ 5.0% Enrichment) 400ppm soluble boron			
nominal (10.906")	maximum (8.98")	nominal (5/16")	0.9236±0.0007††
maximum (10.966")	maximum (8.98")	nominal (5/16")	0.9176±0.0008
minimum (10.846")	nominal (8.92")	nominal (5/16")	0.9227±0.0010
minimum (10.846")	minimum (8.86")	nominal (5/16")	0.9159±0.0008
nominal (10.906")	nominal-0.04" (8.88")	nom.+0.05" (0.3625")	0.9232±0.0009
nominal (10.906")	nominal (8.92")	nominal (5/16")	0.9158±0.0007
MPC-32 (17x17A @ 5.0% Enrichment) 2200 ppm soluble boron†††			
minimum (9.158")	minimum (8.69")	nominal (9/32")	0.9399±0.0007
nominal (9.218")	nominal (8.75")	nominal (9/32")	0.9370±0.0007
maximum (9.278")	maximum (8.81")	nominal (9/32")	0.9313±0.0008
nominal+0.05" (9.268")	nominal (8.75")	nominal+0.05" (0.331")	0.9356±0.0007
minimum+0.05"(9.208")	minimum (8.69")	nominal+0.05" (0.331")	0.9395±0.0008
maximum (9.278")	Maximum-0.05" (8.76")	nominal+0.05" (0.331")	0.9330±0.0008

Notes:

1. Values in parentheses are the actual value used.

† Tolerance for pitch and box I.D. are ± 0.06".
Tolerance for box wall thickness is +0.05", -0.00".

†† Numbers are 1σ statistical uncertainties.

††† for 0.075" sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.

6.4.4.2.1 Bounding Intact Assemblies

Intact BWR assemblies stored together with DFCs are limited to a maximum planar average enrichment of 3.7 wt% ^{235}U, regardless of the fuel class. The results presented in Table 6.1.7 are for different enrichments for each class, ranging between 2.7 and 4.2 wt% ^{235}U, making it difficult to identify the bounding assembly. Therefore, additional calculations were performed for the bounding assembly in each assembly class with a planar average enrichment of 3.7 wt%. The results are summarized in Table 6.4.7 and demonstrate that the assembly classes 9x9E and 9x9F have the highest reactivity. These two classes share the same bounding assembly (see footnotes for Tables 6.2.33 and 6.2.34 for further details). This bounding assembly is used as the intact BWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-24E are limited to a maximum enrichment of 4.0 wt% ^{235}U without credit for soluble boron and to a maximum enrichment of 5.0 wt% with credit for soluble boron, regardless of the fuel class. The results presented in Table 6.1.3 are for different enrichments for each class, ranging between 4.2 and 5.0 wt% ^{235}U, making it difficult to directly identify the bounding assembly. However, Table 6.1.4 shows results for an enrichment of 5.0 wt% for all fuel classes, with a soluble boron concentration of 300 ppm. The assembly class 15x15H has the highest reactivity. This is consistent with the results in Table 6.1.3, where the assembly class 15x15H is among the classes with the highest reactivity, but has the lowest initial enrichment. Therefore, in the MPC-24E, the 15x15H assembly is used as the intact PWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-32 are limited to a maximum enrichment of 5.0 wt%, regardless of the fuel class. Table 6.1.5 and Table 6.1.6 show results for enrichments of 4.1 wt% and 5.0 wt%, respectively, for all fuel classes. Since different minimum soluble boron concentrations are used for different groups of assembly classes, the assembly class with the highest reactivity in each group is used as the intact assembly for the calculations with DFCs in the MPC-32. These assembly classes are

- 14x14C for all 14x14 assembly classes;
- 15x15B for assembly classes 15x15A, B, C, G and I;
- 15x15F for assembly classes 15x15D, E, F and H;
- 16x16A for assembly classes 16x16A, B and C;
- 17x17A assembly class and
- 17x17C for assembly classes 17x17B and C.

voided guide tubes, a water density of 1.0 g/cc in the DFC and MPC, 2900 ppm soluble boron, and an enrichment of 5.0 wt% ^{235}U for the intact and damaged fuel and fuel debris. For this case, results are summarized in Table 6.4.13. For each condition, the table lists the highest maximum k_{eff}, including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the fuel pellet diameter in the DFC has an insignificant effect on reactivity, and that reactivity decreases with decreasing water density. The latter demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. Therefore, a typical fuel pellet diameter and a water density of 1.0 in the DFCs are used for all further analyses. Two enrichment levels are analyzed, 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, consistent with the analyses for intact fuel only. In any calculation, the same enrichment is used for the intact fuel and the damaged fuel and fuel debris. For both enrichment levels, analyses are performed with voided and filled guide tubes, each with water densities of 0.93 and 1.0 g/cm^3 in the MPC. In all cases, the water density inside the DFCs is assumed to be 1.0 g/cm^3, since this is the most reactive condition as shown in Table 6.4.13. Results are summarized in Table 6.4.14. For each group of assembly classes, the table shows the soluble boron level and the highest maximum k_{eff} for the various moderation conditions of the intact assembly. The highest maximum k_{eff} is the highest value of any of the hypothetical fuel debris configurations, i.e. various arrays of bare fuel rods. All maximum k_{eff} values are below the 0.95 regulatory limit. Conditions of damaged fuel such as assemblies with missing rods or collapsed assemblies were not analyzed in the MPC-32, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods.

6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods. The number of dummy rods that is used to replace missing and/or damaged fuel rods is not limited.

6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.15. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in ^{235}U (equivalent to UO_2 fuel with an enrichment of approximately up to 1.7 wt% ^{235}U), the large spacing between the rods (the pitch is approximately 1", the cladding OD is 0.412") and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in Tables 6.1.7 and 6.1.8 this result demonstrates, that the k_{eff} for a Thoria Rod

Canister loaded into the MPC-68 or the MPC-68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of k_{eff} < 0.95. *The evaluation presented here was performed with fresh fuel, but for a condition where all basket locations were assumed to be filled with a Thoria Rod canister, while only a single canister is permitted. The conversion of Th-232 to U-233 during depletion may results in a slight increase in reactivity for this hypothetical case of a basket entirely filled with Thoria Rod Canisters, however, the real condition of a single canister loaded together with spent fuel would still be bounded by the design basis case with fuel assemblies only.*

6.4.7 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware such as Thimble Plugs (TPs), Burnable Poison Rod Assemblies (BPRAs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and similar devices are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged. This conclusion is supported by the calculation listed in Table 6.2.4, which shows a significant reduction in reactivity as a result of voided guide tubes, i.e. the removal of the water from the guide tubes.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes

Table 6.4.10

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES
FOR THE MPC-32 AT 5.0 wt% ENRICHMENT

Fuel Class	**Minimum Soluble Boron Content (ppm)**	**MPC-32 @ 5.0 %**			
		Guide Tubes Filled,		**Guide Tubes Voided,**	
		1.0 g/cm^3	**0.93 g/cm^3**	**1.0 g/cm^3**	**0.93 g/cm^3**
14x14A	1900	0.8984	0.9000	0.8953	0.8943
14x14B	1900	0.9210	0.9214	0.9164	0.9118
14x14C	1900	0.9371	0.9376	0.9480	0.9421
14x14D	1900	0.9050	0.9027	0.8947	0.8904
14x14E	1900	0.7415	0.7301	n/a	n/a
15x15A	2500	0.9210	0.9223	0.9230	0.9210
15x15B	2500	0.9402	0.9420	0.9429	0.9421
15x15C	2500	0.9258	0.9292	0.9307	0.9293
15x15D	2600	0.9426	0.9419	0.9466	0.9440
15x15E	2600	0.9394	0.9415	0.9434	0.9442
15x15F	2600	0.9445	0.9465	0.9483	0.9460
15x15G	2500	0.9228	0.9244	0.9251	0.9243
15X15H	2600	0.9271	0.9301	0.9317	0.9333
15x15I‡	2500	0.9402	0.9363	-	-
16X16A	2000	0.9377	0.9375	0.9429	0.9389
16X16B	2000	0.9326	0.9338	0.9378	0.9358
16X16C	2000	0.9208	0.9193	0.9091	0.9055
17x17A	2200	0.9472	0.9468	0.9475	0.9459
17x17B	2600	0.9345	0.9358	0.9371	0.9356
17X17C	2600	0.9417	0.9431	0.9437	0.9430

‡ This array/class has solid guide rods that cannot be filled or voided.

Table 6.4.11

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES FOR THE MPC-32 AT 4.1 wt% ENRICHMENT

Fuel Class	**Minimum Soluble Boron Content (ppm)**	**MPC-32 @ 4.1 %**			
		Guide Tubes Filled		**Guide Tubes Voided**	
		1.0 g/cm^3	**0.93 g/cm^3**	**1.0 g/cm^3**	**0.93 g/cm^3**
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
14x14E	1300	0.7340	0.7204	n/a	n/a
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
15x15I§	1800	0.9340	0.9316	-	-
16X16A	1400	0.9367	0.9347	0.9375	0.9308
16X16B	1400	0.9336	0.9319	0.9354	0.9283
16X16C	1400	0.9178	0.9130	0.9039	0.8965
17x17A	1600	0.9421	0.9413	0.9396	0.9357
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

§ This array/class has solid guide rods that cannot be filled or voided.

Table 6.4.14

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-32
WITH UP TO 8 DFCs UNDER VARIOUS MODERATION CONDITIONS.

Fuel Assembly Class of Intact Fuel	**Initial Enrichment (wt% ^{235}U)**	**Minimum Soluble Boron Content (ppm)**	**Maximum k_{eff}**			
			Filled Guide Tubes		**Voided Guide Tubes**	
			1.0 g/cm^3	**0.93 g/cm^3**	**1.0 g/cm^3**	**0.93 g/cm^3**
14x14A through 14x14E	4.1	1500	0.9277	0.9283	0.9336	0.9298
	5.0	2300	0.9139	0.9180	0.9269	0.9262
15x15A, B, C, G, I	4.1	1900	0.9345	0.9350	0.9350	0.9326
	5.0	2700	0.9307	0.9346	0.9347	0.9365
15x15D, E, F, H	4.1	2100	0.9322	0.9336	0.9340	0.9329
	5.0	2900	0.9342	0.9375	0.9385	0.9397
16x16A, B, C	4.1	1500	0.9330	0.9332	0.9348	0.9333
	5.0	2300	0.9212	0.9246	0.9283	0.9299
17x17A	4.1	1800	0.9298	0.9310	0.9311	0.9283
	5.0	2600	0.9228	0.9274	0.9273	0.9298
17x17B, C	4.1	2100	0.9284	0.9290	0.9294	0.9285
	5.0	2900	0.9308	0.9338	0.9355	0.9367

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, 4.1% Enrichment, Bounding Cases					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.9041	0.9001	0.0006	0.3185
B14x14B01	HI-STAR	0.9257	0.9216	0.0007	0.4049
14x14C01	HI-STAR	0.9423	0.9382	0.0007	0.4862
14x14D01	HI-STAR	0.8970	0.8931	0.0006	0.5474
14x14E02	HI-STAR	0.7340	0.7300	0.0006	0.6817
15x15A01	HI-STAR	0.9206	0.9167	0.0006	0.5072
B15x15B01	HI-STAR	0.9397	0.9358	0.0006	0.4566
B15x15C01	HI-STAR	0.9266	0.9227	0.0006	0.4167
15x15D04	HI-STAR	0.9384	0.9345	0.0006	0.5594
15x15E01	HI-STAR	0.9365	0.9326	0.0006	0.5403
15x15F01	HI-STORM (DRY)	0.4691	0.4658	0.0003	1.207E+04
15x15F01	HI-TRAC	0.9403	0.9364	0.0006	0.4938
15x15F01	HI-STAR	0.9411	0.9371	0.0006	0.4923
15x15G01	HI-STAR	0.9147	0.9108	0.0006	0.5880
15x15H01	HI-STAR	0.9276	0.9237	0.0006	0.4710
15x15I01	HI-STAR	0.9340	0.9301	0.0006	0.5488
16x16A03	HI-STAR	0.9375	0.9333	0.0007	0.4488
16x16B01	HI-STAR	0.9354	0.9315	0.0006	0.4253
16x16C01	HI-STAR	0.9178	0.9137	0.0007	0.4408
17x17A01	HI-STAR	0.9421	0.9381	0.0006	0.3534
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, 5.0% Enrichment, Bounding Cases					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
16x16A03	HI-STAR	0.9429	0.9388	0.0007	0.5920
16x16B01	HI-STAR	0.9378	0.9339	0.0006	0.5632
16x16C01	HI-STAR	0.9208	0.9167	0.0007	0.5898
17x17A01	HI-STAR	0.9475	0.9435	0.0006	0.5285
17x17B06	HI-STAR	0.9371	0.9331	0.0006	0.6705
17x17C02	HI-TRAC	0.9436	0.9396	0.0006	0.6773
17x17C02	HI-STAR	0.9437	0.9399	0.0006	0.6780

Note: Maximum k_{eff} = Calculated k_{eff} + $K_c \times \sigma_c$ + Bias + σ_B
where:
K_c = 2.0
σ_c = Std. Dev. (1-sigma)
Bias = 0.0021
σ_B = 0.0006
See Subsection 6.4.3 for further explanation.

SUPPLEMENT 6.III[1]: CRITICALITY EVALUATION OF THE MPC-68M

6.III.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.III.1.1], 10CFR72.124 [6.III.1.2], and NUREG-0800 Section 9.1.2 [6.III.1.3], the results in this supplement demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM 100 System with the MPC-68M, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Criticality safety of the HI-STORM 100 System with the MPC-68M depends on the following principal design parameters:

- The inherent geometry of the fuel basket design of the MPC-68M;

- The incorporation of spatially distributed B-10 isotope in the Metamic-HT fuel basket structure. Based on the tests for the neutron absorber content in Metamic-HT (see Appendix 1.III.A and Supplement 9.III), and consistent with the approach taken for Metamic (see Section 9.1.5.3.2), 90% of the minimum B-10 (B_4C) content is credited in the analysis. With a specified minimum B_4C content of 10 wt%, the concentration credited in the analysis is therefore 9 wt%.

The off-normal and accident conditions defined in Section 2.2 are applicable to the HI-STORM System using the MPC-68M. These accidents are considered in Supplement 11.III and have no adverse effect on the design parameters important to criticality safety, except for the non-mechanistic tip-over event, which could result in limited plastic deformation of the basket. However, a bounding basket deformation is already included in the criticality models for normal conditions, and thus, from the criticality safety standpoint, the off-normal and accident conditions are identical to those for normal conditions.

Results of the design basis criticality safety calculations for a single internally flooded HI-TRAC transfer cask with full water reflection on all sides (limiting cases for the HI-STORM 100 System), loaded with undamaged fuel assemblies are listed in Table 6.III.1.1, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.III.3), and including the calculational bias, uncertainties, and calculational statistics. Table 6.III.1.1 provides the information for undamaged fuel without known or suspected cladding defects larger than pinhole leaks or hairline cracks, while Table 6.III.1.4 provides information for low-enriched, channeled BWR undamaged fuel without known or suspected grossly breached fuel rods. In addition, a result for a single internally dry (no moderator) HI-STORM storage cask with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, is listed in Table 6.III.1.2 to confirm the low reactivity of the HI-STORM 100 System with an MPC-68M in storage. The maximum k_{eff} for an MPC-68M loaded with up to 16 DFCs is listed in Table 6.III.1.3.

[1] Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.III.1.4].

TABLE 6.III.1.1

BOUNDING MAXIMUM keff VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68M (HI-TRAC 100)

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% U-235)	Maximum keff
7x7B	4.8	0.9243
8x8B	4.8	0.9294
8x8C	4.8	0.9302
8x8D	4.8	0.9307
8x8E	4.8	0.9211
8x8F	4.5	0.9245
9x9A	4.8	0.9341
9x9B	4.8	0.9330
9x9C	4.8	0.9254
9x9D	4.8	0.9254
9x9E/F	4.5	0.9254
9x9G	4.8	0.9211
10x10A	4.8	0.9360
10x10B	4.8	0.9353
10x10C	4.8	0.9321
10x10F	4.7	0.9356
10x10G	4.75	0.9472

Note: The results presented in the table above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

TABLE 6.III.1.2

REPRESENTATIVE k_{eff} VALUES FOR MPC-68M IN THE HI-STORM 100 OVERPACK

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
10x10A	4.8	0.3754

TABLE 6.III.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68M WITH UP TO 16 DFCs

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
All BWR Classes except 8x8F, 9x9E/F, 10x10F and 10x10G	4.8	0.9408
8x8F and 9x9E/F	4.0	0.9028
10x10F and 10x10G	4.6	0.9453

TABLE 6.III.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68M WITH LOW ENRICHED, CHANNELED BWR FUEL

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
All BWR Classes	3.3	0.9269

Note: The results presented in Tables 6.III.1.2, 6.III.1.3 and 6.III.1.4 above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

6.III.2 SPENT FUEL LOADING

The BWR fuel assembly classes/arrays which are authorized for the MPC-68 are qualified for the MPC-68M, except for the 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D and 10x10E. Additionally, the MPC-68M is qualified for two new assembly classes, 10x10F and 10x10G. Information on those classes is provided in Supplement 2.III, Table 2.III.3. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for all other fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt% ^{235}U are authorized in the MPC-68M.

BWR assemblies are specified in the Table 2.1.4 and Table 2.III.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in Appendix 6.B for the MPC-68 show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum k_{eff}. To confirm this for the higher enrichments analyzed here, additional calculations were performed for the assembly class 10x10A in the MPC-68M, and are presented in Table 6.III.2.1 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt% ^{235}U is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt% ^{235}U is close to the maximum rod enrichment of 5.0 wt% ^{235}U, the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt% ^{235}U, while the remainder of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results in Table 6.III.2.1 indicate that there may be a potential positive reactivity effect (+0.0021) due to distributed enrichments. Therefore, additional studies with distributed enrichments were performed and are presented in Table 6.III.2.2. These include all cases from Appendix B (for 8x8 and 9x9 assembly types), now evaluated in the MPC-68M, and additional cases for the 10x10G which has the highest reactivity of all assembly classes. The cases from Appendix B show no statistically significant increase, and in most cases a decrease in reactivity as a result of the distributed enrichment. Nevertheless, for conservatism an additional bias of 0.0021 is applied to the results for all 10x10 fuel assembly classes in Section 6.III.1, including the cases with damaged fuel and low enriched channeled fuel. Note that for the studies presented in the remainder of this supplement this bias is not included since those studies focus on reactivity differences rather than absolute values of k_{eff}.

TABLE 6.III.2.2

ADDITIONAL CALCULATIONS OF THE REACTIVITY EFFECT OF DISTRIBUTED ENRICHMENTS IN BWR FUEL IN THE MPC-68M

Assembly Class	Enrichment	Maximum k_{eff}	Description	Delta-k_{eff}
8x8C	4.8	0.8273	Average Enrichment	-0.0044
8x8C	4.8	0.8229	Distributed Enrichment	
8x8C	4.8	0.8876	Average Enrichment	-0.0040
8x8C	4.8	0.8836	Distributed Enrichment	
8x8D	4.8	0.8550	Average Enrichment	+0.0004
8x8D	4.8	0.8554	Distributed Enrichment	
8x8D	4.8	0.8774	Average Enrichment	-0.0017
8x8D	4.8	0.8757	Distributed Enrichment	
8x8D	4.8	0.8855	Average Enrichment	-0.0026
8x8D	4.8	0.8829	Distributed Enrichment	
9x9B	4.8	0.9103	Average Enrichment	-0.0023
9x9B	4.8	0.9080	Distributed Enrichment	
9x9D	4.8	0.8467	Average Enrichment	-0.0095
9x9D	4.8	0.8372	Distributed Enrichment	
8x8C	4.8	0.9023	Average Enrichment	-0.0025
8x8C	4.8	0.8998	Distributed Enrichment	
8x8C	4.8	0.9165	Average Enrichment	-0.0003
8x8C	4.8	0.9162	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0253
10x10G	4.75	0.9198	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0268
10x10G	4.75	0.9183	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0009
10x10G	4.75	0.9442	Distributed Enrichment	
10x10G	4.75	0.9451	Average Enrichment	-0.0010
10x10G	4.75	0.9441	Distributed Enrichment	

TABLE 6.III.3.1

EVALUATION OF BASKET MANUFACTURING TOLERANCES FOR MPC-68M

Box I.D.	Box Wall Thickness	Maximum keff
10x10A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9307
increased (6.07")	minimum (0.38")	0.9288
minimum (5.99")	minimum (0.38")	0.9334
minimum, including deformation (5.96")	minimum (0.38")	**0.9339**
7x7B, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9154
nominal (6.05")	minimum (0.38")	0.9196
minimum, including deformation (5.96")	minimum (0.38")	**0.9243**
8x8D, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9230
nominal (6.05")	minimum (0.38")	0.9265
minimum, including deformation (5.96")	minimum (0.38")	**0.9307**
9x9A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9301
minimum, including deformation (5.96")	minimum (0.38")	**0.9341**
10x10G, 4.75% Enrichment		
nominal (6.05")	nominal (0.40")	0.9365
nominal (6.05")	minimum (0.38")	0.9409
minimum, including deformation (5.96")	minimum (0.38")	**0.9451**
10x10A, 4.8% Enrichment, Damaged Fuel		
nominal (6.05")	nominal (0.40")	0.9316
nominal (6.05")	minimum (0.38")	0.9348
minimum, including deformation (5.96")	minimum (0.38")	**0.9387**

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

6.III.4 CRITICALITY CALCULATIONS

The calculations in this supplement use the same computer codes and methodologies that are used in the main part of Chapter 6. Specifically, the conservative approach to model damaged fuel and fuel debris, using arrays of bare fuel rods, is the same (see discussion in Subsection 6.III.4.1 below).

The basket design of the MPC-68M is essentially identical to that of the MPC-68, in respect to the characteristics important to criticality safety. Specifically,

- The number and configuration of the cells for undamaged and damaged fuel/fuel debris are unchanged;
- The basket dimensions are essentially the same; and
- The same poison material (B_4C) is used, but a larger ^{10}B content in the basket walls.

The content is also the same, except for the following

- Higher enrichments are qualified, consistent with the higher ^{10}B content in the basket walls; and
- Two additional fuel assembly types are analyzed, that are variations of existing types with slightly different dimension.

To verify that the bounding fuel parameter variations analyzed in the MPC-68 are also applicable to the MPC-68M, additional studies are performed and discussed in Subsection 6.III.4.2 below.

Due to the strong similarity in the basket design, the conclusions of the various studies presented in the main part of this Chapter on the MPC-68 are directly applicable to the MPC-68M. Nevertheless, to confirm this is also applicable to the MPC-68M, numerous studies with various moderation conditions that conclude that the fully flooded basket is the bounding case are re-analyzed and discussed in subsection 6.III.4.3. All analyzes are therefore performed under the following condition:

- Basket, and DFCs as applicable, are fully flooded with pure water at the maximum density; and
- Pellet-to-clad gaps of undamaged assemblies are assumed flooded (see also discussion in Subsection 6.III.4.2 below)
- All assemblies and DFCs are located eccentrically in the basket, closest to the center of the basket.

Results for all design basis calculations are listed in Subsection 6.III.1. All maximum k_{eff} values are below the regulatory limit of 0.95.

6.III.4.1 Damaged Fuel and Fuel Debris

For damaged fuel and fuel debris in the MPC-68M the same conservative approach is used as in the main part of this chapter, see discussion in Section 6.4.4, specifically 6.4.4.2. Important aspects of this approach that ensure its conservatisms are as follows:

- All damaged fuel and fuel debris must be in damaged fuel containers (DFCs), and located in specifically designated cells on the periphery of the basket as specified in Table 2.1.22.
- A conservative model is used that bounds both damaged fuel and fuel debris. In other words, damaged fuel is always conservatively modeled as fuel debris.
- The model consists of regular arrays of fuel rods without cladding. The rods pitch (array size) is varied to determine the optimum moderation condition.
- Undamaged and damaged fuel/fuel debris in the same basket have the same enrichment limit, which may be different from the enrichment limit for undamaged fuel only.
- The results for loading with undamaged fuel only in Table 6.III.1.1 utilize different enrichment limits for different assembly classes, to ensure that the maximum k_{eff} is always below 0.95. It is therefore not possible to establish a single bounding assembly class/enrichment combination to be used in all analyzes with damaged fuel/fuel debris. Therefore, and in order to optimize the enrichment for the loading of undamaged and damaged fuel/fuel debris for each assembly class, undamaged assemblies are grouped by enrichment limit, and the undamaged assembly with the highest maximum k_{eff} in each group is used for the calculations together with damaged fuel/fuel debris. These are:
    - Undamaged assemblies of 4.5 wt%: Assembly class 9x9E/F. For the calculations with undamaged and damaged fuel, an enrichment of 4.0 wt% is used.
    - Undamaged assembly of 4.7 and 4.75 wt%: Assembly class 10x10G. For the calculations with undamaged and damaged fuel, an enrichment of 4.6 wt% is used.
    - Undamaged assembly of 4.8 wt%: Assembly class 10x10A. For the calculations with undamaged and damaged fuel, an enrichment of 4.8 wt% is used.
- Consistent with the results in the main part of this chapter for the MPC-68, array sizes of 9x9, 10x10 and 11x11 show the optimum moderation condition. This is confirmed for undamaged assembly classes 9x9E/F, 10x10A and 10x10G by evaluating all arrays from 3x3 to 17x17 rods.

6.III.4.2 Fuel Parameters and Parameter Variations

In the main part of the FSAR, extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- maximum fuel rod pitch,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only)
- part length rods (if present) removed.

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely

1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:

- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.
- Fuel assemblies are predominantly characterized by the amount of fuel and the fuel-to-water (moderator) ratio. Increasing the amount of fuel, or the enrichment of the fuel, will increase the amount of fissile material, and therefore increase reactivity. Regarding the fuel-to-water ratio, it is important to note that commercial BWR assemblies are undermoderated, i.e. they do not contain enough water for a maximum possible reactivity.
- The neutron poison in the basket walls uses B-10, which is an absorber of thermal neutrons. This poison therefore also needs water (moderator) to be effective. This places a specific importance on the amount of water between the outer rows of the fuel assemblies and the basket cell walls. Note that this explains some of the differences in reactivity between the different assembly types in the same basket, even for the same enrichment, where assemblies with a smaller cross section, i.e. which have more water between the periphery of the assembly and the surrounding wall, generally have a lower reactivity.

Based on these characteristics, the following conclusions can be made:

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536, a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not by a positive increase in reactivity from the reduction in

demonstrated that the phenomenon of a peak in reactivity at low moderator densities (sometimes called "optimum" moderation) does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. All calculations are therefore performed with full water density inside the MPCs.

- Partial Flooding: The partial flooding of the basket, either in horizontal or vertical direction, reduces the amount of fuel that partakes effectively in the thermal fission process, while essentially maintaining the fuel-to-water ratio in the volume that is still flooded. This will therefore result in a reduction of the reactivity of the system (similar to that of the reduction of the active length), and due to the similarity of the fuel assemblies is not dependent on the specific fuel type. The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cm^3) water and the remainder of the cask is filled with steam consisting of ordinary water at a low partial density (0.002 g/cm^3 or less), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.III.4.8. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. Note that the studies for partial flooding are performed with the design basis model for the assembly class 10x10A that has the partial length rods removed for added conservatism, while the calculations in the main part of the chapter for the MPC-68 were performed for an assembly class that did not include partial length rods. This shows that the conclusion from partial flooding, i.e. that the fully flooded condition is bounding, applies equally to assemblies with and without partial lengths rods.

- Pellet-to-clad Gap Flooding: As demonstrated by the studies shown in Table 6.III.4.2, all assemblies are undermoderated. Flooding the pellet-to-clad gap will therefore improve the moderation and therefore increase reactivity for all assembly types.

- Preferential Flooding: The only preferential flooding situation that may be credible is the flooding of the bottom section of the DFCs while the rest of the MPC internal cavity is already drained. In this condition, the undamaged assemblies have a negligible effect on the system reactivity since they are not flooded with water. The dominating effect is from the damaged fuel model in the DFCs. However, the damaged fuel model is conservatively based on an optimum moderated array of bare fuel rods in water, and therefore representative of all fuel types and therefore the fully flooded condition is bounding of the preferential flooding condition.

6.III.4.4 Low Enriched, Channeled BWR fuel

The calculations in this subsection show that low enriched, channeled BWR fuel with indeterminable cladding condition is acceptable for loading in the storage locations of the MPC-68M without placing those fuel assemblies into DFCs, hence classifying those assemblies as undamaged. The main characteristics that must be assured are:

- The channel is present and attached to the fuel assembly in the standard fashion; and
- The channel is essentially undamaged; and
- The maximum planar average enrichment of the assembly is less than or equal to 3.3 wt% ^{235}U

This analysis covers older assemblies, where the cladding integrity is uncertain, and where a verification of the cladding condition is prohibitive. An example of this type of fuel is the so-called CILC (Copper Induced Localized Corrosion) fuel, which has potential corrosion-induced damaged to the cladding but does not have grossly breached spent fuel rods.

The presence of the essentially undamaged and attached channel confines the fuel rods to a limited volume and the low enrichment, limits the reactivity of the fuel even under optimum moderation conditions. Due to the uncertain cladding condition, the analysis of this fuel follows essentially the same approach as that for the Damaged Fuel and Fuel Debris, i.e. bare fuel rod arrays of varying sizes are analyzed within the confines of the channel. This is an extremely conservative modeling approach for this condition, since reconfiguration is not expected and cladding would still be present.

Calculations are performed with these assemblies in all cells of the MPC-68M, without DFCs. The results of this conservative analysis are listed in Table 6.III.4.9 and show that the system remains below the regulatory limit.

In addition, calculations are performed for the MPC-68M with checkerboard configuration of normal undamaged fuel and low enriched, channeled BWR fuel fuel without DFCs. The results of this analysis are listed in Table 6.III.4.10 and show that the reactivity remains below the regulatory limit and bounded by the reference undamaged fuel assembly in all cells.

These results confirm that even with unknown cladding condition the maximum k_{eff} values are below the regulatory limit when fully flooded and loaded with any of the BWR candidate fuel assemblies, therefore if the cladding is not grossly breached and the fuel assembly is structurally sound it can be considered undamaged when loaded in an MPC-68M.

6.III.4.5 Thoria Rod Canister

The criticality evaluation of thoria rod canister was performed for MPC-68 or MPC-68F and results presented in Section 6.4.6 show that it is permissible to load the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. While only a single canister is qualified for storage, the analysis assumes such a canister in every basket cell, and calculates a very low reactivity of less than 0.2 for this condition, based on a UO2 content of 1.8 wt%. *The conversion of Th-232 to U-233 during depletion may results in a slight increase in reactivity for the hypothetical case of a MPC-68 or MPC-68F entirely filled with Thoria Rod Canisters, however, the real condition of a single canister loaded together with spent fuel would still be bounded by the design basis case with fuel assemblies only.* Since the MPC-68M has equal or better criticality performance than the MPC-68 due to the basket itself being made from the neutron absorber, Metamic-HT. Without any further evaluations it can therefore be concluded that, from a criticality perspective, the thoria rods with the actual composition can be safely stored in the HI-STORM 100 system in an MPC-68M canister.

TABLE 6.III.4.1

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH UNDAMAGED AND DAMAGED FUEL/FUEL DEBRIS

	Maximum k_{eff},		
Bare Rod Array inside the DFC	**Assembly Classes 8x8F and 9x9E/F (4.0 wt%)**	**Assembly Class 10x10F and 10x10G (4.6 wt%)**	**All other assembly classes (4.8 wt%)**
3x3	0.8926	0.9310	0.9267
6x6	0.8942	0.9338	0.9295
8x8	0.8986	0.9395	0.9344
9x9	0.9028	0.9414	0.9371
10x10	0.9024	0.9432	0.9387
11x11	0.9024	0.9420	0.9381
12x12	0.9018	0.9412	0.9373
13x13	0.9007	0.9397	0.9353
14x14	0.8993	0.9385	0.9352
16x16	0.8985	0.9376	0.9335
17x17	0.8976	0.9366	0.9328

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.2

MAXIMUM k_{eff} VALUES IN THE MPC-68M FOR VARIOUS FUEL TYPES WITH VOIDED PELLET TO CLAD GAP

Assembly Classes	Enrichment	Maximum k_{eff} (Voided Gap)	Reference k_{eff} (Flooded Gap)	Delta k_{eff}
7X7B	4.8	0.9185	0.9243	-0.0058
8x8B	4.8	0.9210	0.9294	-0.0084
8x8C	4.8	0.9243	0.9302	-0.0059
8x8D	4.8	0.9245	0.9307	-0.0062
8x8E	4.8	0.9152	0.9211	-0.0059
8x8F	4.5	0.9191	0.9245	-0.0054
9x9A	4.8	0.9290	0.9341	-0.0051
9x9B	4.8	0.9202	0.9330	-0.0128
9x9C	4.8	0.9203	0.9254	-0.0051
9x9D	4.8	0.9210	0.9254	-0.0044
9x9E	4.5	0.9157	0.9254	-0.0097
9x9G	4.8	0.9160	0.9211	-0.0051
10x10A	4.8	0.9311	0.9339	-0.0028
10x10B	4.8	0.9242	0.9332	-0.0090
10x10C	4.8	0.9253	0.9300	-0.0047
10x10F	4.7	0.9301	0.9335	-0.0034
10x10G	4.75	0.9403	0.9451	-0.0048

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.4

MAXIMUM k_{eff} VALUES IN THE MPC-68M
FOR VARIOUS PART LENGTH ROD LENGTHS (10x10G, 4.75% Enrichment)

Maximum k_{eff}	**Description**
0.9193	Full Length Rods Only
0.9279	Part Length Rods 25% length
0.9377	Part Length Rods 50% length
0.9439	Part Length Rods 75% length
0.9451	All Rods

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.7

MAXIMUM k_{eff} VALUES IN THE MPC-68M
FOR EXTERNAL FLOODING

Internal Water Density (%)	External Water Density (%)	7x7B (4.8%)	8x8F (4.5%)	9x9C (4.8%)	10x10A (4.8%)	10x10G (4.75%)
100	100	0.9243	0.9245	0.9254	0.9351	0.9451
100	70	0.9238	0.9250	0.9259	0.9353	0.9450
100	50	0.9235	0.9239	0.9249	0.9336	0.9456
100	20	0.9234	0.9245	0.9259	0.9342	0.9452
100	10	0.9234	0.9245	0.9257	0.9351	0.9446
100	05	0.9238	0.9247	0.9258	0.9346	0.9458
100	01	0.9230	0.9256	0.9261	0.9341	0.9459

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

TABLE 6.III.4.9

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED BWR FUEL IN ALL CELLS

Rod Array inside the Channel	Maximum k_{eff}
3x3	0.2045
6x6	0.7229
8x8	0.8900
9x9	0.9219
10x10	0.9248
11x11	0.9065
12x12	0.8689
13x13	0.8161
14x14	0.7562
16x16	0.6653
17x17	0.6449

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.10

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH MIXTURE OF UNDAMAGED BWR FUEL AND LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED BWR FUEL

Configuration	Rod Array	10x10A, 4.8 wt% ^{235}U		10x10G, 4.75 wt% ^{235}U	
		Maximum k_{eff}	Reactivity Effect	Maximum k_{eff}	Reactivity Effect
Undamaged Normal Fuel in all Cells	-	0.9339	Reference	0.9451	Reference
Checkerboard of CILC Fuel at 3.3 wt% ^{235}U and Undamaged Fuel	3x3	0.6218	-0.3121	0.6247	-0.3204
	6x6	0.8241	-0.1098	0.8281	-0.1170
	8x8	0.9110	-0.0229	0.9161	-0.0290
	9x9	0.9275	-0.0064	0.9329	-0.0122
	10x10	0.9297	-0.0042	0.9341	-0.0110
	11x11	0.9206	-0.0133	0.9264	-0.0187
	12x12	0.9054	-0.0285	0.9109	-0.0342
	13x13	0.8865	-0.0474	0.8913	-0.0538
	14x14	0.8666	-0.0673	0.8719	-0.0732
	16x16	0.8514	-0.0825	0.8563	-0.0888
	17x17	0.8505	-0.0834	0.8561	-0.0890

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

6.III.5 CRITICALITY BENCHMARK EXPERIMENTS

Same as in Section 6.5

6.III.6 REGULATORY COMPLIANCE

Same as in Section 6.6

6.III.7 REFERENCES

[6.III.1.1] NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington, D.C., January 1997.

[6.III.1.2] 10CFR72.124, "Criteria for Nuclear Criticality Safety."

[6.III.1.3] USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 2 - July 1981.

[6.III.1.4] "HI-STAR 100 AND HI-STORM 100 ADDITIONAL CRITICALITY CALCULATIONS", Holtec Report HI-2012771 Rev.20 (proprietary)

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 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 Section 2.1.9 have been selected for loading into the MPC.

Caution:
In accordance with the definition of "Undamaged Fuel," some low-enriched channeled fuel must be shown to be without known or suspected grossly breached spent fuel rods. This determination can be made based on review of records, fuel sipping, or other method.

8.2 ISFSI OPERATIONS

The HI-STORM 100 System heat removal system is a totally passive system. Maintenance on the HI-STORM system is typically limited to cleaning and touch-up painting of the overpacks, repair and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM system. These include, but are not limited to, wind storms, heavy snow storms, fires inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM overpack, clear the debris, check the cask pitch, and/or replace damaged vent screens), perform the corrective action, verify that the system is operable (check ventilation flow paths and radiation). In the event of significant damage to the HI-STORM, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM overpack. If necessary, the procedures in Section 8.1 may be used to reposition a HI-STORM overpack for minor repairs and maintenance. In extreme cases, Section 8.3 may be used as guidance for unloading the MPC from the HI-STORM.

Note:
The heat removal system operability surveillance involves performing a visual examination on the HI-STORM exit and inlet vent screens to ensure that the vents remain clear or verifying the temperature rise from ambient to outlet is within prescribed limits. The metallic vent screens if damaged may allow leaves, debris or animals to enter the duct and block the flow of air to the MPC.

ALARA Warning:
Operators should practice ALARA principals when inspecting the vent screens. In most cases, binoculars allow the operator to perform the surveillance from a low dose area.

8.2.1 Perform the heat removal operability surveillance.

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

8.2.2 ISFSI Security Operations shall be performed in accordance with the approved site security program plan.

compliance with 10CFR72.104[10.0.1] and 10CFR20[10.1.1] taking into account the actual site boundary distance and fuel characteristics.

Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

10.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Section 11.1, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and off-normal handling of HI-TRAC) do not result in the degradation of the HI-STORM 100 System shielding effectiveness. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that of normal conditions.

10.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 10.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 11 presents the results of the evaluations performed to demonstrate that the HI-STORM 100 System can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [10.0.1]. The accident events addressed in Chapter 11 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, confinement boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, partial blockage of MPC basket air inlets, and 100% blockage of air inlets.

The worst-case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-STORM overpack is the 30 day 100% blockage of air inlets event. A bounding volume of the neutron shield layer (concrete) is assumed to reach temperatures of at least 350°F in the cask body and lid. Material composition and density may be affected and therefore, the shielding effectiveness of this volume of concrete is degraded. However, even when considering worst case conservatisms as discussed in Section 5.1.2, the dose rate at the controlled area boundary is bounded by the HI-TRAC loss of water in water jacket accident and does not exceed 10CFR72.106 limits.

from the bottom of the overpack, the minimum distance to the controlled area boundary, was calculated for the HI-STORM 100S Version B with an MPC-24 for an assumed accident duration of 30 days. The burnup and cooling time of the fuel was 60,000 MWD/MTU and 3 years, which is more conservative than consistent with the off-site dose analysis reported in Chapter 11, Table 11.4.1 and the . This combination of overpack, MPC, burnup and cooling time is the same as that used in Chapters 5 and 10 for off-site dose calculations. The results presented below demonstrate that the regulatory requirements of 10CFR72.106 are easily met.

Distance	Dose Rate (mrem/hr)	Accident Duration	Total Dose (mrem)	10CFR72.106 Limit (mrem)
100 meters	2.36	720 hours or 30 days	1699.2	5000

11.2.3.4 Tip-Over Accident Corrective Action

Following a tip-over accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the structural damage shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. If determined necessary, the MPC shall be returned to the facility for fuel unloading or transferred to either a HI-STAR or HI-STORM overpack in accordance with Chapter 8 for a duration that is determined to be appropriate. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs are required and will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.4 Fire Accident

11.2.4.1 Cause of Fire

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low, a conservative fire has been assumed and analyzed. The analysis shows that the HI-STORM 100 System continues to perform its structural, confinement, thermal, and subcriticality functions.

11.2.13 100% Blockage of Air Inlets

11.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all four bottom inlets. Such blockage of the inlets may be postulated to occur as a result of a flood, blizzard snow accumulation, tornado debris, or volcanic activity.

11.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet ducts is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass, and correspondingly large thermal capacity of the storage overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for a number of days. This accident condition is, however, a short duration event that will be identified and corrected by scheduled periodic surveillance at the ISFSI site depending on the cask heat load at the time of inspection. The temperature rise due to this accident event is small for heat loads much lower than design maximum heat load even if the condition persists for a substantial number of days. As evaluated in Sub-section 4.6.2.4, mandatory 30-day surveillance of casks is required under heat loads less than or equal to the threshold heat load specified in Table 4.6.8 at the time of inspection.

Structural

There are no structural consequences as a result of this event. However, since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, structural evaluation of a missile impact coincident with the 100% vent blockage event is evaluated in Section 3.4.8.1 to demonstrate safety of the system.

Thermal

A thermal analysis is performed in Subsection 4.6.2 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and component temperatures remain below their temperature limits. The MPC internal pressure for this event is evaluated in Subsection 4.6.2 and is bounded by the design basis internal pressure for accident conditions (Table 2.2.1).

Since the mandatory surveillance frequency for MPCs at or below threshold decay heat is substantial, additional thermal evaluations are performed in Section 4.6.2.5 to demonstrate that the

MPCs are safe at off-normal and accident conditions coincident with the 100% vent blockage event at threshold heat load.

Shielding

For a short duration event, 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.

For the duration of a 30-day event, this accident results in an increase in the radiation dose rates at 100 m. The shielding evaluation presented in Chapter 5 demonstrates 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

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100 System, if the blockage is removed in the specified time period.

11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets 32-hour 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.

For the 100% blockage of air inlets 30-day accident, the material composition and density of portion of the concrete of the HI-STORM 100 System may be affected. 100 m dose rates will increase as a result of this event. Shielding analysis documented in Section 5.1.2 shows that the resulting dose rate is bounded by the HI-TRAC loss of water in water jacket accident condition and therefore is in compliance with the 10CFR72.106 limits.

11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within 32 hours for cask heat loads greater than that specified in Table 4.6.8 at the time of inspection. For cask containing MPCs with total heat load and per cell decay heat less than or equal to threshold heat load (Table 4.6.8), blockage is cleared within 30 days. Upon detection of the complete blockage of the air inlet ducts, the ISFSI operator shall assign personnel to clear the blockage with mechanical and manual means as necessary. After clearing the overpack ducts, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air ducts of a cask with heat loads greater than the threshold heat load (Table 4.6.8) at the time of occurrence for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to demonstrate adequate heat removal for the duration of the event. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and Overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain below their short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet ducts and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet ducts using pumps or fire-hoses or blowing air into the air outlet ducts using fans, to directly cool the MPC.

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(s) from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM 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

b. Canister material mechanical properties for structural integrity of the confinement boundary.

c. Canister and basket material thermal properties and dimensions for heat transfer control.

d. Canister and basket material composition and dimensions for dose rate control.

12.2.9 HI-STORM Overpack/VVM

a HI-STORM overpack/VVM material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations, as applicable.

b. HI-STORM overpack/VVM material thermal properties and dimensions for heat transfer control.

c. HI-STORM overpack/VVM material composition and dimensions for dose rate control.

12.2.10 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below execute the approach and equations described in Section 2.1.9.1 for determining allowable decay heat per storage location, burnup, and cooling time for the approved cask contents.

Example 1

In this example, a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern and will be stored in an aboveground HI-STORM system.

Step 1: Pick a value of X between 0.5 and 3. For this example X will be 2.8.

Step 2: Calculate $q_{Region2}$ as described in Section 2.1.9.1.2:

$$q_{Region2} = (2 \text{ x } 34)/[(1 + (2.8)^{0.2075}) \text{ x } ((12 \text{ x } 2.8) + 20)] = 0.5668 \text{ kW}^{\dagger}$$

Step 3: Calculate $q_{Region1}$ as described in Section 2.1.9.1.2:

$$q_{Region1} = X \text{ x } q_{Region2} = 2.8 \text{ x } 0.5668 = 1.5871 \text{ kW}$$

† Results are arbitrarily rounded to four decimal places.

Step 4: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 as described in Section 2.1.9.1.3.

Table 12.2.3 provides three hypothetical fuel assemblies in the 16x16A array/class that will be evaluated for acceptability for loading in the MPC-32 example above. The decay heat values in Table 12.2.3 are calculated by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is not acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.1 and 12.1.2. The solution is to develop another table using an enrichment of 3.0 wt.% ^{235}U or less to determine this fuel assembly's suitability for loading in this MPC-32.

Fuel Assembly Number 2 is not acceptable for loading unless a unique maximum allowable burnup for a cooling time of 3.3 years is calculated by linear interpolation between the values in Table 12.2.1 for 3 years and 4 years of cooling. Linear interpolation yields a maximum burnup of 36,497 MWD/MTU (rounded down from 36,497.2), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.

Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment that those in Table 12.2.1, which is still below the actual initial enrichment of Fuel Assembly Number 3. Due to its relatively low total decay heat of 0.5 kW (fuel: 0.4, non-fuel hardware: 0.1), Fuel Assembly Number 3 may be stored in Region 1 or Region 2.

Example 2

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern in an aboveground system. Assuming the same value 'X', the same maximum fuel storage location decay heats are calculated. The equation in Section 2.1.9.1.3 is executed for each fuel assembly using its exact initial enrichment to determine its maximum allowable burnup. Linear interpolation is used to further refine the maximum allowable burnup value between cooling times, if necessary.

Fuel Assembly Number 1: The calculated allowable burnup for 3.0 wt.% ^{235}U and a decay heat value of 1.5871 kW ($q_{region1}$) is 44,905 MWD/MTU at 4 years minimum cooling. Its decay heat is too high for loading in Region 2. Comparing the fuel assembly burnup and total decay heat of

the contents† (fuel (1.01 kW) plus non-fuel hardware (0.5 kW)) to the calculated limits indicates that the fuel assembly, including the non-fuel hardware, is acceptable for storage in Region 1.

Fuel Assembly Number 2: The calculated allowable burnup for 3.2 wt.% ^{235}U and a decay heat value of 1.5871 kW ($q_{region1}$) is 32,989 MWD/MTU for 3 years cooling and 45,382 MWD/MTU for 4 years cooling. Linearly interpolating between these values for a cooling time of 3.3 years yields a maximum allowable burnup of 36,706 MWD/MTU and, therefore, the assembly is acceptable for storage in Region 1. This fuel assembly's decay heat is also too high for loading in Region 2.

Fuel Assembly Number 3: The calculated allowable maximum burnup for 4.3 wt.% ^{235}U and a decay heat value of 0.5668 ($q_{Region2}$) is 41,693 MWD/MTU for 18 years cooling. Comparing the fuel assembly burnup and total decay heat of the contents (fuel plus non-fuel hardware) against the calculated limits indicates that the fuel assembly and non-fuel hardware are acceptable for storage. Therefore, the assembly is acceptable for storage in Region 2. This fuel assembly would also be acceptable for loading in Region 1 (this conclusion is inferred, but not demonstrated).

Example 3

In this example, a demonstration of the use of burnup versus cooling time tables for uniform fuel loading is provided. In this example it will be assumed that the MPC-68 is being loaded with array/class 9x9A fuel and will be stored in an aboveground HI-STORM system.

Step 1: CoC TS Appendix B Table 2.4-1 provides the heat load limit on each storage location (q_{max}). For MPC-68 this is 0.5 kW.

Step 2: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments if the fuel being loaded varies significantly in initial enrichment. It is conservative to choose the lowest value of initial enrichment to generate the table.

In this example, two enrichments will be used: 3.0 and 4.5. Tables 12.2.4 and 12.2.5 show the burnup versus cooling time tables calculated for these enrichments for the respective q_{max}.

Table 12.2.6 provides three hypothetical fuel assemblies in the 9x9A array/class that will be evaluated for acceptability for loading in the MPC-68 example above. The decay heat values in Table 12.2.6 would be calculated by the user. The other information would be taken from the fuel assembly and reactor operating records.

All of the assemblies meet the per cell heat load limit of 0.5 kW.

† The assumption is made that the non-fuel hardware meets burnup and cooling time limits in Table 2.1.25.

Fuel Assembly Number 1 is acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.4 and the burnup is lower than that allowed for the cooling time of the assembly.

Fuel Assembly Number 2 is not acceptable for loading based on the current tables. The fuel assembly burnup is greater than allowed by Table 12.2.4, even with linear interpolation (30978 MWD/MTU). Fuel Assembly Number 2 may be acceptable for loading if a new table is created specifically for an initial enrichment of 3.5 wt% and the allowable burnup is greater than 35250.

Fuel Assembly Number 3 is acceptable for loading based on the allowable burnups in Table 12.2.5.

12.2.11 Verifying Compliance with Total MPC Heat Load

Some operational steps and/or use of particular equipment are required if Q_{CoC} is above a certain value, e.g. 28.74 kW in the MPC-32. These include supplemental cooling, forced helium dehydration, helium backfill pressure, and surveillance requirements for LCO 3.1.2. These examples demonstrate the logic behind the decisions for these operational steps. Time to boil limits and vacuum drying are also considered in these examples.

Example 1:

Table 12.2.7 contains a proposed heat load pattern for loading a MPC-68 into an aboveground HI-STORM 100 System. The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are ≤ 0.50 kW (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 18.917 kW.
4. The maximum heat load in any cell is 0.460 kW.
5. $Q_{CoC,}$ as defined in Section 2.1.9.1.2 equation c is 31.280 kW.

Recommendations based on the general observations <u>without further site-specific analysis</u>:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying because the Q_{CoC} heat load is greater than 30 kW (See FSAR Table 4.5.1).
2. Forced Helium Dehydration: The MPC should be dried using forced helium

dehydration since the Q_{CoC} heat load exceeds the vacuum drying threshold heat loads (See FSAR Table 4.5.1).

3. Helium Backfill Pressure Range: The MPC should be backfilled to the higher pressure range given in the TS because the Q_{CoC} heat load exceeds the threshold heat loads in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system would be required for on-site transport of High Burnup Fuel in the HI-TRAC after the MPC is dried, backfilled and sealed because the Q_{CoC} heat load exceeds the 90% design basis threshold heat load in FSAR Table 4.5.4.
5. Heat Removal Surveillance (LCO 3.1.2): The user has 24 hours to clear blockage on the system containing this MPC since the Q_{CoC} heat load (assuming the pattern is at the time of inspection) exceeds the 28.152 kW (=0.414 kW*68) threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 18.917 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 18.917 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

Example 2

Table 12.2.8 contains a proposed heat load pattern for loading a MPC-32. The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are ≤ 1.062 kW (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 17.471 kW.
4. The maximum heat load in any cell is 0.826 kW.
5. $Q_{CoC,}$ as defined in Section 2.1.9.1.2 equation c is 26.432 kW.

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC can be dried using vacuum drying since the Q_{CoC} heat load is bounded by the threshold heat load Q2 in FSAR Table 4.5.1. The vacuum

drying is time limited as Q_{CoC} exceeds threshold heat load Q1 in FSAR Table 4.5.1.

2. Forced Helium Dehydration: The MPC can be dried using forced helium dehydration but it is not required.
3. Helium Backfill Pressure Range: The MPC may be backfilled to either pressure range given in the TS because the Q_{CoC} heat load is bounded by the threshold heat load in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system would NOT be required for on-site transport in the HI-TRAC after the MPC is dried, backfilled and sealed because the Q_{CoC} heat load is bounded by the 90% design basis threshold heat load in FSAR Table 4.5.4.
5. Heat Removal Surveillance (LCO 3.1.2): The user has 64 hours to clear blockage on the system containing this MPC since the Q_{CoC} heat load(assuming the pattern is at the time of inspection) is bounded by the 28.74 kW threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 17.471 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 17.471 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

Example 3

Table 12.2.9 contains a proposed heat load pattern for loading a MPC-32. The table provides the decay heat of each storage location. It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.

General observations on this loading plan:

1. The heat loads do not meet the CoC limits for Uniform Loading, i.e. some cells are ≥ 1.0625 kW (See Table 2.1.26).
2. The X value that most closely meets this pattern (See Table 2.1.30) is 1.5 which means the inner locations cannot have a total decay heat greater than 1.282 kW and the outer locations cannot have a total decay heat greater than 0.855 kW. Note that the pattern also meets the criteria for any X value ≥ 1.5.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 20.697 kW.
4. The maximum heat load in any cell is 1.273 kW.
5. Since this MPC is loaded in a regionalized pattern, $Q_{CoC,}$ as defined in Section 2.1.9.1.2 equation e is 32.484 kW. (12*1.282+20*0.855)

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying since the Q_{CoC} heat load under uniform loading (1.273 kWx32 equals 40.736 kW) exceeds the threshold heat loads in FSAR Table 4.5.1.
2. Forced Helium Dehydration: The MPC must be dried using forced helium dehydration only because vacuum drying is not permitted (see above) and regionalized loading Q_{CoC} is bounded by the design basis heat load in FSAR Table 4.5.1.
3. Helium Backfill Pressure Range: The MPC must be backfilled to the higher pressure range given in the TS because the uniform loading Q_{CoC} heat load exceeds the threshold heat load in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system is required for on-site transport of High Burnup Fuel in the HI-TRAC after the MPC is dried, backfilled and sealed because both uniform loading Q_{CoC} and storage cell heat loads under regionalized storage exceed the 90% design basis threshold heat load in FSAR Table 4.5.4.
5. Heat Removal Surveillance (LCO 3.1.2): The user has 24 hours to clear blockage on the system containing this MPC since the uniform loading Q_{CoC} heat load (assuming the pattern is at the time of inspection) exceeds the 28.74 kW threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 20.697 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 20.697 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

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)
($q_{Region\ 1}$ = 1.5871 kW, $q_{Region\ 2}$ = 0.5668 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	34797	11101
≥4	47590	17870
≥5	56438	23272
≥6	62533	27157
≥7	66963	29907
≥8	68200	31935
≥9	68200	33510
≥10	68200	34785
≥11	68200	35927
≥12	68200	36894
≥13	68200	37790
≥14	68200	38593
≥15	68200	39419
≥16	68200	40191
≥17	68200	40937
≥18	68200	41643
≥19	68200	42363
≥20	68200	43094

Table 12.2.3

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 16x16A)

FUEL ASSEMBLY NUMBER	ENRICHMENT (wt. % ^{235}U)	FUEL ASSEMBLY BURNUP (MWD/MTU)	FUEL ASSEMBLY COOLING TIME (years)	FUEL ASSEMBLY DECAY HEAT (kW)	NON-FUEL HARDWARE STORED WITH ASSEMBLY	NFH DECAY HEAT (kW)
1	3.0	37100	4.7	1.01	BPRA	0.5
2	3.2	35250	3.3	1.45	NA	NA
3	4.3	41276	18.2	0.4	BPRA	0.1

Table 12.2.4

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-68, Array/Class 9x9A, and Enrichment = 3.0 wt.% ^{235}U)
(q_{max} = 0.5 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP (MWD/MTU)
≥3	27739
≥4	38536
≥5	46268
≥6	51583
≥7	55424
≥8	58303
≥9	60733
≥10	62798
≥11	64609
≥12	66331
≥13	68005
≥14	68200
≥15	68200
≥16	68200
≥17	68200
≥18	68200
≥19	68200
≥20	68200

Table 12.2.5

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-68, Array/Class 9x9A, and Enrichment =4.5 wt.% ^{235}U)
(q_{max} = 0.5 kW)

MINIMUM COOLING TIME (years)	**MAXIMUM ALLOWABLE BURNUP (MWD/MTU)**
≥3	30017
≥4	41399
≥5	49359
≥6	54839
≥7	58856
≥8	61932
≥9	64534
≥10	66802
≥11	68200
≥12	68200
≥13	68200
≥14	68200
≥15	68200
≥16	68200
≥17	68200
≥18	68200
≥19	68200
≥20	68200

Table 12.2.6

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 9x9A)

FUEL ASSEMBLY NUMBER	**ENRICHMENT (wt. % ^{235}U)**	**FUEL ASSEMBLY BURNUP (MWD/MTU)**	**FUEL ASSEMBLY COOLING TIME (years)**	**FUEL ASSEMBLY DECAY HEAT (kW)**
1	3.0	37100	4.7	0.3
2	3.5	35250	3.3	0.495
3	4.5	41276	18.2	0.2

BASES

ACTIONS (continued)

B.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours for OVERPACKS containing MPCs with heat loads in excess of the heat loads in Table B.1-1 (below) at the time of inspection. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

Table B.1-1
(Threshold* heat loads for HI-STORM 100 System Surveillance Frequency and Completion Time to restore heat removal system to operable status)

MPC Model(s)	**Threshold Heat Load (per canister)**	**Threshold Heat Load (per assembly)**
24 (all variants)	18 kW	0.75 kW
68 (all variants)	18 kW	0..264 kW
32 (all variants)	16 kW	0.5 kW

Alternatively, for OVERPACKS containing MPCs heat loads up to the thresholds in Table B.1-1 at the time of inspection, the system must be restored to operable status within twenty four hours. Twenty four hours is a reasonable period of time for these lower heat load systems since the temperature limits of the system components and fuel cladding are not exceeded and the event is not time limiting.

(continued)

BASES

ACTIONS (continued)

C.1

For MPCs with heat loads greater than the thresholds in Table B.1-1, if the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the cask user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.

For MPCs with heat loads less than or equal to the threshold heat loads in Table B.1-1, there will be inconsequential temperature increase to the OVERPACK concrete if the system is not restored to operable status within 24 hours. If the heat removal system cannot be restored to operable status within 24 hours, the same actions as above are required.

C.2.1

In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented.

(continued)

BASES

ACTIONS

C.2.1 (continued)

This Required Action must be complete in 64 hours (after entering Condition C) for an aboveground system with an MPC decay heat load of 28.74 kW or less, in 24 hours (after entering Condition C) for an aboveground system with an MPC decay heat load greater than 28.74 kW, and in 16 hours for an underground system. These Completion Times are consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 72, 32 or 24 hours after event initiation, respectively. For MPC heat loads up to the thresholds in Table B.1-1, system components temperatures do not exceed their 30 day accident temperature limits.

The Completion Time reflects the 8 or 24 hours to complete Required Action B.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all air ducts becoming simultaneously blocked by trash or debris.

C.2.2

In lieu of implementing Required Action C.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK. In this case, the requirements of CoC Appendix A, LCO 3.1.4 apply.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2 (continued)

The Frequency of 24 hours for aboveground systems with heat loads that exceed the thresholds in Table B.1-1 at the time of inspection, and 16 hours for underground systems is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts. For aboveground systems containing MPCs with heat loads less than or equal to the threshold heat loads in Table B.1-1 at the time of inspection, the surveillance frequency of 30 days is appropriate, since the system components and peak cladding temperature limits for 30 day accident are not exceeded and the event is not time limiting.

REFERENCES

1. FSAR Chapter 4
2. FSAR Sections 11.2.13 and 11.2.14
3. ANSI/ANS 57.9-1992