CHAPTER 4† THERMAL EVALUATION

4.0 OVERVIEW

The HI-STORM System is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. An array of HI-STORM Systems laid out in a rectilinear pattern will be stored on a concrete ISFSI pad in an open environment. In this section, compliance of the HI-STORM thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI is established. The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. Effects of incident solar radiation (insolation) and partial radiation blockage due to the presence of neighboring casks at an ISFSI site are included in the analyses. Finally, the thermal margins of safety for long-term storage of both moderate burnup (up to 45,000 MWD/MTU) and high burnup spent nuclear fuel (greater than 45,000 MWD/MTU) in the HI-STORM 100 system are quantified. Safe thermal performance during on-site loading, unloading and transfer operations utilizing the HI-TRAC transfer cask is also demonstrated.

The HI-STORM thermal evaluation follows the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4] to demonstrate thermal compliance of the HI-STORM system. . These guidelines provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF)* and other confinement boundary components, and on the maximum permissible pressure in the confinement space under certain operating scenarios. Specifically, the requirements are:

1. The fuel cladding temperature for long-term storage shall be limited to 752°F (400°C).

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

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

4. The maximum internal pressure of the MPC should remain within its design

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1). This chapter has been substantially re-written in support of LAR #3 to improve clarity and to incorporate the 3-D thermal model. Because of extensive editing a clean chapter is issued with this amendment.

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

pressures for normal, off-normal, and accident conditions.

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

5. For fuel assemblies proposed for storage, the cask system should ensure a very low probability of cladding breach during long-term storage.

6. The HI-STORM System should be passively cooled.

7. The thermal performance of the cask shall be in compliance with the design criteria specified in FSAR Chapters 1 and 2 for normal, off-normal, and accident conditions.

As demonstrated in this chapter, the HI-STORM System is designed to comply with all of the criteria listed above. Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions. All thermal analyses to evaluate normal conditions of storage in a HI-STORM storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC transfer cask are described in Section 4.5. All thermal analyses to evaluate off-normal and accident conditions are described in Section 4.6. This FSAR chapter is in full compliance with ISG-11 and with NUREG-1536 guidelines, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.3.

The HI-STORM thermal evaluations for CSF are grouped in two categories of fuel assemblies. The two groups are classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE group of fuel assemblies are characterized by low burnup, long cooling time, and short active fuel lengths. Consequently, their heat loads are dwarfed by the DB group of fuel assemblies. All Dresden-1 (6x6 and 8x8 and a thoria rod canister constituted as part of an 8x8 fuel assembly), Quad+, Humboldt Bay (7x7 and 6x6), Indian Point, Haddam Neck and all stainless-steel clad fuel assemblies are classified as LHE fuel. The low heat emitting characteristics of these fuel assemblies render them non-governing for thermal evaluation. The HI-STORM System temperatures for MPCs loaded with LHE fuel are bounded by design basis evaluations reported in this chapter.

The HI-STORM System is evaluated for two fuel storage scenarios. In one scenario, designated as uniform loading, every basket cell is assumed to be occupied with fuel producing heat at the maximum rate. As discussed in Chapter 2, this storage specification is extremely conservative, and virtually impossible to realize in actual practice. A less unrealistic, yet conservative idealization of storage scenario, designated as regionalized loading, involves defining two discrete regions within the basket. The two regions are designated as Region 1 (inner region) and Region 2 (outer region). Regionalized storage is designed to recognize storage of fuel assemblies having wide disparity in heat emission rates. For further discussion of regionalized storage, Section 2.1 of Chapter 2 should be consulted.

The HI-STORM System is designed for one reference storage condition defined in Table 4.0.1. This condition establishes the required helium backfill pressures computed later in this chapter (See Subsection 4.4.5.1). Having defined the helium backfill pressures an array of analyses are performed to evaluate the range of storage configurations specified in Chapter 2 and results reported in Section 4.4.

Table 4.0.1

REFERENCE HI-STORM OPERATING CONDITIONS

Condition	**Value**
MPC Decay Heat	Table 2.1.26
MPC Operating Pressure	7 atm (absolute)
Normal Ambient Temperature	Table 2.2.2

4.1 DISCUSSION

The HI-STORM FSAR seeks to establish complete compliance with the provisions of ISG-11 [4.1.4]. For this purpose the HI-STORM normal storage fuel cladding temperatures are required to meet the 752°F (400°C) temperature limit for all CSF (See Section 4.3). Additionally, when the MPCs are deployed for storing High Burnup Fuel (HBF) further restrictions during fuel loading activities are set forth to preclude fuel temperatures from exceeding the normal temperature limits. To ensure explicit compliance, a specific term "short term operations" is defined in Chapter 2 to cover all fuel loading activities. ISG-11 fuel cladding temperature limits are applied for short-term operations (see Table 4.3.1).

Potential thermally challenging states for the spent fuel arise if the fuel drying process utilizes the pressure reduction process (i.e., vacuum drying) or when the loaded MPC is inside the HI-TRAC transfer cask. In the latter state, the rate of heat rejection from the MPC is somewhat less compared to the normal storage condition when the MPC is inside the ventilated overpack. Heat dissipation within the MPC is, however, similar to HI-STORM storage as the principal HI-TRAC transfer cask operations with pressurized helium (FHD drying, on-site transport and HI-STORM overpack transfer) are performed in the vertical orientation which preserves MPC thermosiphon action. In the aggregate, the fuel cladding temperatures in the HI-TRAC are modestly higher than the HI-STORM storage temperatures. The short-term evolutions that may be thermally limiting and warrant analysis are:

- i. Vacuum Drying
- ii. Loaded MPC in HI-TRAC in the Vertical Orientation

The threshold MPC heat generation rate at which the HI-STORM peak cladding temperature reaches a steady state equilibrium value approaching the normal storage peak clad temperature limit is computed in this chapter. Likewise, the MPC heat generation rates that produce the steady state equilibrium temperature approaching the normal storage peak clad temperature limit for the MPC in HI-TRAC are computed in this chapter. These computed heat generation rates directly bear upon the compliance of the system with ISG-11 [4.1.4] and are, accordingly, adopted in the system Technical Specifications for high burnup fuel (HBF).

The aboveground HI-STORM system consists of a sealed MPC situated inside a vertically-oriented, ventilated storage overpack. Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top, respectively, of the cylindrical overpack. The SNF assemblies reside inside the MPC, which is sealed with a welded lid to form the confinement boundary. The MPC contains a stainless-steel honeycomb fuel basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. Each fuel basket panel, with the exception of exterior panels on the MPC-68 and MPC inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM System to the environment by passive heat transport mechanisms only.

The helium backfill gas plays an important role in the MPC's thermal performance. The helium fills all the spaces between solid components and provides an improved conduction medium (compared to air) for dissipating decay heat in the MPC. Within the MPC the pressurized helium environment sustains a closed loop thermosiphon action, removing SNF heat by an upward flow of helium through the storage cells. This MPC internal convection heat dissipation mechanism is illustrated in Figure 4.1.1. On the outside of the MPC a ducted overpack construction with a vertical annulus facilitates an upward flow of air by buoyancy forces. The annulus ventilation flow cools the hot MPC surfaces and safely transports heat to the outside environment. The annulus ventilation cooling mechanism is illustrated in Figure 4.1.2. To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC confinement boundary is designed and fabricated to comply with the provisions of the ASME B&PV Code Section III, Subsection NB (to the maximum extent practical), as an all-seal-welded pressure vessel with redundant closures. It is demonstrated in Section 11.1.3 that the failure of one field-welded pressure boundary seal will not result in a breach of the pressure boundary. The helium gas is therefore assumed to be retained in an undiluted state, and may be credited in the thermal analyses.

An important thermal design criterion imposed on the HI-STORM System is to limit the maximum fuel cladding temperature to within design basis limits (Table 4.3.1) for long-term storage of design basis SNF assemblies. An equally important requirement is to minimize temperature gradients in the MPC so as to minimize thermal stresses. In order to meet these design objectives, the MPC baskets are designed to possess certain distinctive characteristics, which are summarized in the following.

The MPC design minimizes resistance to heat transfer within the basket and basket periphery regions. This is ensured by an uninterrupted panel-to-panel connectivity realized in the all-welded honeycomb basket structure. The MPC design incorporates top and bottom plenums with interconnected downcomer paths. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket, and by elongated semicircular holes in each basket cell wall. The bottom plenum is formed by large elongated semicircular holes at the base of all cell walls. The MPC basket is designed to eliminate structural discontinuities (i.e., gaps) which introduce added thermal resistances to heat flow. Consequently, temperature gradients are minimized in the design, which results in lower thermal stresses within the basket. Low thermal stresses are also ensured by an MPC design that permits unrestrained axial and radial growth of the basket. The possibility of stresses due to restraint on basket periphery thermal growth is eliminated by providing adequate basket-to-canister shell gaps to allow for basket thermal growth during all operational modes.

The MPCs design maximum decay heat loads for storage of zircaloy clad fuel are listed in Table 4.0.1. Storage of stainless steel clad fuel is permitted for a low decay heat limit set forth in Chapter 2 (Tables 2.1.17 through 2.1.24). Storage of zircaloy clad fuel with stainless steel clad fuel in an MPC is permitted. In this scenario, the zircaloy clad fuel must meet the lower decay heat limits for stainless steel clad fuel. The axial heat distribution in each fuel assembly is assumed to follow the burnup profiles set forth in Table 2.1.11.

The HI-STORM thermal analysis is performed on the FLUENT [4.1.2] Computational Fluid Dynamics (CFD) program. To ensure a high degree of confidence in the HI-STORM thermal evaluations, the modeling of principal heat dissipation mechanisms – MPC internal convection and annulus ventilation cooling (See Figures 4.1.1 and 4.1.2) – are benchmarked using data from tests conducted with casks loaded with irradiated SNF. The benchmark work is archived in QA validated Holtec reports identified in Table 4.1.1. Additionally, in Section 4.4, a benchmarking study of the porous media model used to represent the hydraulic resistances of a fuel storage cell, is provided.

Table 4.1.1

Benchmarking of HI-STORM Thermal Models

Benchmark Model	**Tested Cask**	**Test Reference**	**Holtec Benchmark Report**
MPC Internal Convection	TN-24P	EPRI [4.1.3]	HI-992252 [4.1.5]
Overpack Ventilation Cooling	VSC-17	EPRI [4.1.7]	HI-2043258 [4.1.6]

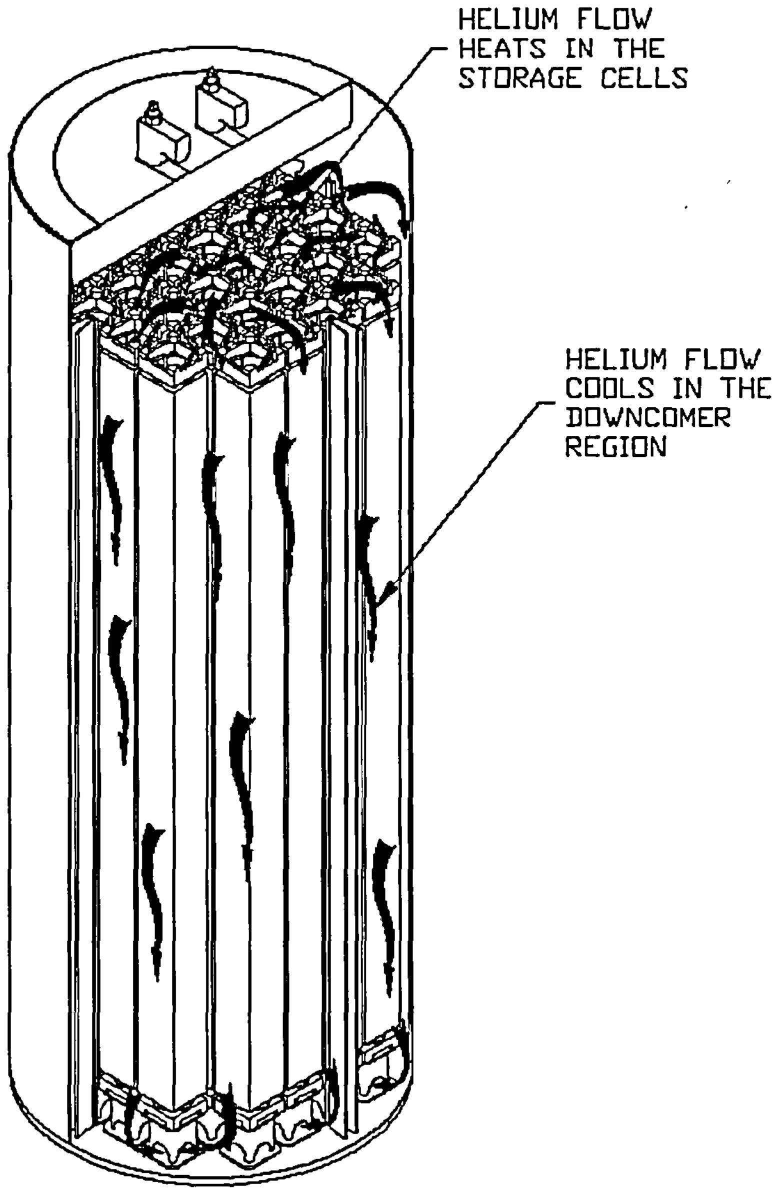

FIGURE 4.1.1: MPC INTERNAL HELIUM CIRCULATION

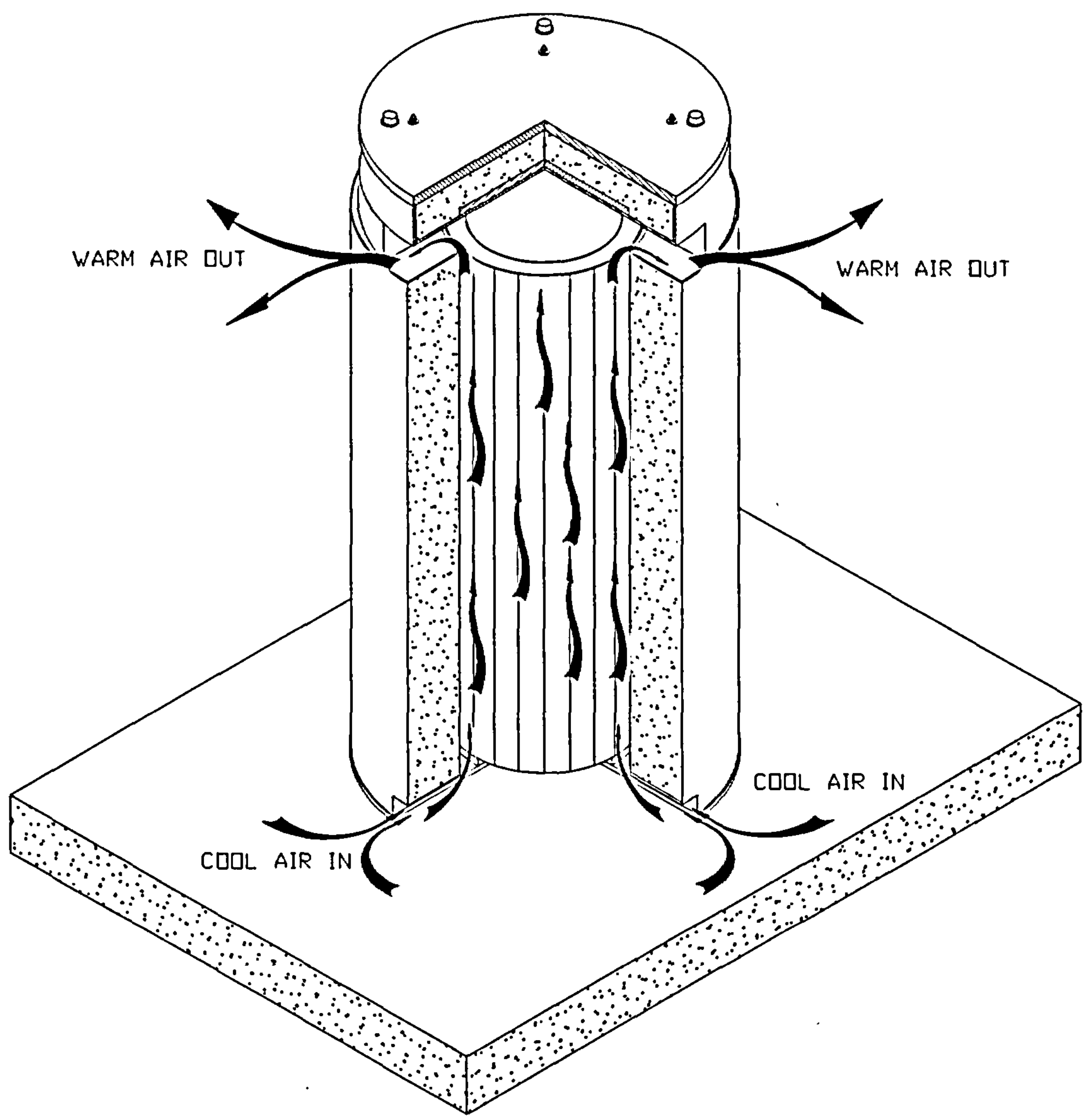

FIGURE 4.1.2: VENTILATION COOLING OF A HI-STORM SYSTEM

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Materials present in the MPCs include stainless steels (Alloy X), neutron absorber (Boral or METAMIC) and helium. Materials present in the HI-STORM storage overpack include carbon steels and concrete. Materials present in the HI-TRAC transfer cask include carbon steel, lead, Holtite-A neutron shield, paints (See Appendix 1.C) and demineralized water. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Individual thermal conductivities of the alloys that comprise the Alloy X materials and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2 and 4.2.3 provide numerical thermal conductivity data of materials at several representative temperatures. The currently approved neutron absorber materials, (Boral™ and Metamic™) are both made of aluminum powder and boron carbide powder. Although their manufacturing processes differ, from a thermal standpoint, their ability to conduct heat is virtually identical. Therefore, the values of conductivity of the original neutron absorber (Boral) continue to be used in the thermal calculations.

For the HI-STORM overpack, the thermal conductivity of concrete and the emissivity/absorptivity of painted surfaces are particularly important. Recognizing the considerable variations in reported values for these properties, the values that are conservative with respect to both authoritative references and values used in analyses on previously licensed cask dockets have been selected. Specific discussions of the conservatism of the selected values are included in the following paragraphs.

As specified in Table 4.2.1, the concrete thermal conductivity is taken from Marks' Standard Handbook for Mechanical Engineers, which is conservative compared to a variety of recognized concrete codes and references. Neville, in his book "Properties of Concrete" (4^{th} Edition, 1996), gives concrete conductivity values as high as 2.1 Btu/(hr×ft×°F). For concrete with siliceous aggregates, the type to be used in HI-STORM overpacks, Neville reports conductivities of at least 1.2 Btu/(hr×ft×°F). Data from Loudon and Stacey, extracted from Neville, reports conductivities of 0.980 to 1.310 Btu/(hr×ft×°F) for normal weight concrete protected from the weather. ACI-207.1R provides thermal conductivity values for seventeen structures (mostly dams) at temperatures from 50-150°F. Every thermal conductivity value reported in ACI-207.1R is greater than the value used in the HI-STORM thermal analyses. Additionally, the NRC has previously approved analyses that use higher conductivity values than those applied in the HI-STORM thermal analysis. For example, thermal calculations for the NRC approved Vectra NUHOMS cask system (June 1996, Rev. 4A) used thermal conductivities as high as 1.17 Btu/(hr×ft×°F) at 100°F. Based on these considerations, the concrete thermal conductivity value chosen for HI-STORM thermal analyses is considered to be conservative.

Holtite-A is a composite material consisting of approximately 37 wt% epoxy polymer, 1 wt% B_4C and 62 wt% aluminum trihydrate. While polymers are generally characterized by a low conductivity (0.05 to 0.2 Btu/ft-hr-°F), the addition of fillers in substantial amounts can raise the mixture conductivity by up to a factor of ten. The thermal conductivity of epoxy filled resins with alumina is reported in the technical literature† as approximately 0.5 Btu/ft-hr-°F and higher. A conservatively postulated conductivity of 0.3 Btu/ft-hr-°F is used in the thermal models for the neutron shield region* (in the HI-TRAC transfer cask). As the thermal inertia of the neutron shield is not credited in the analyses, the density and heat capacity properties are not reported herein.

Surface emissivity data for key materials of construction are provided in Table 4.2.4. The emissivity properties of painted external surfaces are generally excellent. Kern [4.2.5] reports an emissivity range of 0.8 to 0.98 for a wide variety of paints. In the HI-STORM thermal analysis, an emissivity of 0.85†† is applied to painted surfaces. A conservative solar absorptivity coefficient of 1.0 is applied to all exposed overpack surfaces.

In Table 4.2.5, the heat capacity and density of the MPC, overpack and CSF materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. The temperature-dependent values of the viscosities of helium and air are provided in Table 4.2.6.

The heat transfer coefficient for exposed surfaces is calculated by accounting for both natural convection and thermal radiation heat transfer. The natural convection coefficient depends upon the product of Grashof (Gr) and Prandtl (Pr) numbers. Following the approach developed by Jakob and Hawkins [4.2.9], the product Gr×Pr is expressed as $L^3\Delta TZ$, where L is height of the overpack, ΔT is overpack surface temperature differential and Z is a parameter based on air properties, which are known functions of temperature, evaluated at the average film temperature. The temperature-dependent values of Z are provided in Table 4.2.7.

† "Principles of Polymer Systems", F. Rodriguez, Hemisphere Publishing Company (Chapter 10).
* The thermal conductivity value used in the thermal models for the neutron shield region is confirmed to be bounded by the Holtite-A test data [4.2.13] with a margin.
†† This is conservative with respect to prior cask industry practice, which has historically utilized higher emissivities [4.2.16].

Table 4.2.1

SUMMARY OF HI-STORM SYSTEM MATERIALS
THERMAL PROPERTY REFERENCES

Material	Emissivity	Conductivity	Density	Heat Capacity
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	[4.2.3], [4.2.17], [4.2.18], [4.2.7]	NUREG [4.2.6]	Rust [4.2.4]	Rust [4.2.4]
UO_2	Note 1	NUREG [4.2.6]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel (machined forgings)*	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Stainless Steel Plates†	ORNL [4.2.11], [4.2.12]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Carbon Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Boral	Note 1	Test Data (Note 2)	Test Data (Note 2)	Test Data (Note 2)
Holtite-A	Note 1	[4.2.13]	Not Used	Not Used
Concrete	Note 1	Marks' [4.2.1]	Appendix 1.D	Handbook [4.2.2]
Lead	Note 1	Handbook [4.2.2]	Handbook [4.2.2]	Handbook [4.2.2]
Water	Note 1	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
METAMIC	Note 1	Test Data [4.2.14], [4.2.15]	Test Data [4.2.14], [4.2.15]	Test Data [4.2.14], [4.2.15]

Note 1: Emissivity not reported as radiation heat dissipation from these surfaces is conservatively neglected.
Note 2: AAR Structures Boral thermophysical test data.

* Used in the top lid of the MPC.
† Used in the basket panels, neutron absorber sheathing, MPC shell, and MPC baseplate.

Table 4.2.2

SUMMARY OF HI-STORM SYSTEM MATERIALS
THERMAL CONDUCTIVITY DATA

Material	**At 200°F (Btu/ft-hr-°F)**	**At 450°F (Btu/ft-hr-°F)**	**At 700°F (Btu/ft-hr-°F)**	**At 1000°F (Btu/ft-hr-°F)**
Helium	0.0976	0.1289	0.1575	0.1890
Air*	0.0173	0.0225	0.0272	0.0336
Alloy X	8.4	9.8	11.0	12.4
Carbon Steel	24.4	23.9	22.4	20.0
Concrete**	1.05	1.05	1.05	1.05
Lead	19.4	17.9	16.9	N/A
Water	0.392	0.368	N/A	N/A

* At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F at 32°F and 0.0176 Btu/ft-hr-°F at 212°F.

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

Table 4.2.3

SUMMARY OF FUEL ELEMENT COMPONENTS
THERMAL CONDUCTIVITY DATA

Zircaloy Cladding		**Fuel (UO_2)**	
Temperature (°F)	Conductivity (Btu/ft-hr-°F)	Temperature (°F)	Conductivity (Btu/ft-hr-°F)
392	8.28*	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28*

* Lowest values of conductivity used in the thermal analyses for conservatism.

Table 4.2.4

SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA*

Material	Emissivity
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel (machined forgings)	0.36
Stainless Steel Plates	0.587**
Carbon Steel	0.66

* See Table 4.2.1 for cited references.

** Lowerbound value from the cited references in Table 4.2.1.

Table 4.2.5

DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY*

Material	Density (lbm/ft^3)	Heat Capacity (Btu/lbm-°F)
Helium	(Ideal Gas Law)	1.24
Zircaloy	409	0.0728
Fuel (UO_2)	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Boral	154.7	0.13
Concrete	140**	0.156
Lead	710	0.031
Water	62.4	0.999
METAMIC	163.4**	0.22**

* See Table 4.2.1 for cited references.

** Lowerbound values reported for conservatism.

Table 4.2.6

GASES VISCOSITY* VARIATION WITH TEMPERATURE

Temperature (°F)	Helium Viscosity (Micropoise)	Temperature (°F)	Air Viscosity (Micropoise)
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	338.4	246.3
463.0	288.7	567.1	293.0
537.8	299.8	701.6	316.7
737.6	338.8	1078.2	377.6
921.2	373.0	-	-
1126.4	409.3	-	-
* Obtained from Rohsenow and Hartnett [4.2.2].			

Table 4.2.7

VARIATION OF NATURAL CONVECTION PROPERTIES PARAMETER "Z" FOR AIR WITH TEMPERATURE

Temperature (°F)	Z ($ft^{-3}°F^{-1}$)*
40	2.1×10^6
140	9.0×10^5
240	4.6×10^5
340	2.6×10^5
440	1.5×10^5
* Obtained from Jakob and Hawkins [4.2.9]	

4.3 SPECIFICATIONS FOR COMPONENTS

HI-STORM System materials and components designated as "Important to Safety" (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) which warrant special attention are summarized in Table 4.3.1. The neutron shielding ability of Holtite-A neutron shield material used in the HI-TRAC transfer cask is ensured by demonstrating that the material exposure temperatures are maintained below the maximum allowable limit. Long-term integrity of SNF is ensured by the HI-STORM System thermal evaluation which demonstrates that fuel cladding temperatures are maintained below design basis limits. Neutron absorber materials used in MPC baskets for criticality control (made from B_4C and aluminum) are stable in excess of 1000°F†. Accordingly 1000°F is conservatively adopted as the short-term temperature limit for neutron absorber materials. The overpack concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that American Concrete Institute (ACI) guidance on temperature limits (see Appendix 1.D) is followed.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term off-normal and severe hypothetical accident conditions. The inherent mechanical characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of HI-STORM System thermal performance, material temperature limits for long-term normal, short-term operations, and off-normal and accident conditions are provided in Table 4.3.1. In Table 4.3.1, ISG-11 [4.1.4] temperature limits are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

4.3.1 Evaluation of Moderate Burnup Fuel

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.1]. Upon cooling below a threshold temperature (T_p), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature T_p are excessive. For moderate burnup fuel, T_p is conservatively estimated as 350°C [4.3.1]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.1]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted in the HI-STORM FSAR for short-term operations for MBF fueled MPCs (see Table 4.3.1).

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

Table 4.3.1

HI-STORM SYSTEM MATERIAL TEMPERATURE LIMITS†

Material	Normal Long-Term Temperature Limits [°F]	Short-Term Temperature Limits [°F]
CSF cladding (zirconium alloys and stainless steel)	752	Short-Term Operations 752 (HBF) 1058 (MBF) Off-Normal and Accident 1058
Neutron Absorber	800	1000
Holtite-A†††	N/A (Not Used)	350 (Short Term Operations)
Concrete‡	300	350
Water	N/A	307§ (Short Term Operations) N/A (Off-Normal and Accident)

† This table specifies temperature limits for non-ASME Code materials. Temperature limits of ASME Code materials (structural steels) are specified in Table 2.2.3.
††† See Chapter 1, Appendix 1.B.
‡ These values are applicable for concrete in the overpack body, overpack lid and overpack pedestal. As stated in Chapter 1 (Appendix 1.D), these limits are compared to the through-thickness section average temperature.
§ Saturation temperature at HI-TRAC water jacket design pressure specified in Table 2.2.1.

4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

Under long-term storage conditions, the HI-STORM System (i.e., HI-STORM overpack and MPC) thermal evaluation is performed in accordance with the guidelines of NUREG-1536 [4.4.1] with the MPC cavity backfilled with helium. Thermal analysis results for the long-term storage scenarios are obtained and reported in this section.

4.4.1 Thermal Model

The MPC basket design consists of four distinct geometries to hold 24 or 32 PWR, or 68 BWR fuel assemblies. The basket is a matrix of interconnected square compartments designed to hold the fuel assemblies in a vertical position under long term storage conditions. The basket is a honeycomb structure of stainless steel (Alloy X) plates with full-length edge-welded intersections to form an integral basket configuration. All individual cell walls, except outer periphery cell walls in the MPC-68 and MPC-32, are provided with neutron absorber plates sandwiched between the box wall and a stainless steel sheathing plate over the full length of the active fuel region. The neutron absorber plates used in all MPCs are made of an aluminum-based, boron carbide-containing material to provide criticality control, while maximizing heat conduction capabilities.

Thermal analysis of the HI-STORM System is performed for an array of limiting heat load scenarios defined in Chapter 2 for uniform and regionalized fuel loading. While the assumption of limiting heat generation imputes a certain symmetry to the cask thermal problem, it grossly overstates the total heat duty of the system in most casks because it is unlikely that any basket would be loaded with fuel emitting heat at their limiting values (See for example a fuel loading scenario discussed in Section 2.1). The principal attributes of the thermal model are described in the following::

i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange is a highly nonlinear function of surface temperatures.

ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profiles in the fuel assemblies.

iii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the MPC is spatially distributed with the maximum values reached in the central core region.

The design basis decay heat for long-term normal storage is specified in Table 2.2.26. The decay heat is conservatively considered to be non-uniformly distributed over the active fuel length based on the axial burnup distributions specified in Chapter 2 (Table 2.1.11).

The thermal model of the HI-STORM in this FSAR actually consists of two distinct models – a 3-D model and an axisymmetric model. The 3-D model has been prepared that simulates each storage cell and seeks to represent the helium flow in each cell, upper and lower plenum and downcomer regions in a rigorous manner. The axisymmetric model replaces the fuel basket and the stored SNF by a porous cylinder having equivalent thermal-hydraulic properties. Unfortunately, the 3-D model is not suitable for making production runs because of its large size. Therefore, the necessity to construct a computationally efficient model remains. In what follows, both the 3-D model and a computationally efficient axisymmetric model are described. The results from both the models are compared for all MPC types to establish the conservatism of the axisymmetric model, which is subsequently used to perform a multitude of production runs.

Certain aspects of the thermal simulation are common to both 3-D and axisymmetric models. For example, the composite wall of the basket consisting of a sandwich of Alloy X panel, the neutron absorber, and the Alloy X sheathing can be replaced by a single orthotropic metal whose conductivities in the three principal directions are equivalent to those of the sandwich. Further details on equivalent representation of the monometallic wall for the sandwich and other regions of the MPC are provided later in this subsection.

4.4.1.1 <u>3-D Thermal Model</u>

i. Overview

The interior of the MPC is a 3-D array of square shaped cells inside an irregularly shaped basket outline confined inside the cylindrical space of the MPC cavity. To ensure an adequate representation of these features, a 3-D geometric model of the MPC is constructed using the FLUENT CFD code pre-processor [4.1.2]. Other than representing the composite cell walls (made up of Alloy X panels, neutron absorber panels and Alloy X sheathing) by a homogeneous panel with equivalent orthotropic (thru-thickness and parallel plates direction) thermal conductivities, the 3-D model requires no idealizations of the fuel basket structure. Further, since it is clearly impractical to model every fuel rod in every stored fuel assembly explicitly, the cross section bounded by the inside of the storage cell (inside of the fuel channel in the case of BWR MPCs), which surrounds the assemblage of fuel rods and the interstitial helium gas (also called the "rodded region"), is replaced with an "equivalent" square homogeneous section characterized by an effective thermal conductivity. Homogenization of the storage cell cross-section is illustrated in Figure 4.4.1. As the effective conductivity of the rodded region includes radiation heat transfer the conductivities will be a strong function of temperature because radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) rises as the fourth power of absolute temperature. Therefore, in effect, the effective conductivity of the equivalent square section (depending on the coincident temperature) will be different throughout the basket. For thermal-hydraulic simulation, each fuel assembly in its storage cell is represented by an equivalent porous medium. For BWR fuel, the presence of the fuel channel divides the storage cell space into two distinct axial flow regions, namely, the in-channel (rodded) region and the square prismatic annulus region (in the case of PWR fuel this modeling complication does not exist).

ii. Details of the 3-D Model

The 3-D model implemented to analyze the HI-STORM system has the following key attributes:

a. As mentioned above, the composite walls in the fuel basket consisting of the Alloy X structural panels, the aluminum-based neutron absorber, and the Alloy X sheathing, are represented by an orthotropic homogeneous panel of equivalent thermal conductivity in the three principal directions. The in-plane and thru-thickness thermal conductivities of the composite wall are computed using a standard procedure for such shapes with certain conservatisms, as described below.

During fabrication, a uniform normal pressure is applied to each "Box Wall - Neutron Absorber - Sheathing" sandwich in the assembly fixture during welding of the sheathing periphery on the box wall. This ensures adequate surface-to-surface contact between the neutron absorber and the adjacent Alloy X surfaces. The mean coefficient of linear expansion of the neutron absorber is higher than the thermal expansion coefficients of the basket and sheathing materials. Consequently, basket heat-up from the stored SNF will further ensure a tight fit of the neutron absorber plate in the sheathing-to-box pocket. Nevertheless the possible presence of small microscopic gaps due to less than perfect surface-to-surface contact requires consideration of an interfacial contact resistance between the neutron absorber and box-sheathing surfaces. In the thermal analysis a 2 mil neutron absorber to pocket gap has been used. This is conservative as the sandwich is engineered to ensure an essentially no-gap fitup and assembly of the neutron-absorber panels. Furthermore, no credit is taken for radiative heat exchange across the neutron absorber to sheathing or neutron absorber to box wall gaps.

Quite obviously, heat conduction properties of a composite "Box Wall - Neutron Absorber - Sheathing" sandwich in the two principal basket cross sectional directions (i.e., thru-thickness and parallel plates direction) are unequal. In the thru-thickness direction, heat is transported across layers of sheathing, helium-gap, neutron absorber and box wall resistances that are essentially in series Heat conduction in the parallel plates direction, in contrast, is through an array of essentially parallel resistances comprised of these several layers listed above. In this manner the composite walls of the fuel basket storage cells are replaced with a solid wall of equivalent through thickness and parallel plates direction conductivities. Table 4.4.1 provides the values of the conductivities as a function of temperature for the different MPC types. These values are used in the axisymmetric model as well.

b. In the case of a BWR CSF, the fuel bundle and the small surrounding spaces inside the fuel "channel" are replaced by an equivalent porous media having the flow impedance properties computed using a 3-D CFD model documented in [4.4.2]. The space between the BWR fuel channel and the storage cell is represented as an open flow annulus. The fuel channel is also explicitly modeled. The porous medium within the channel space is also referred to as the "rodded region". The fuel assembly is assumed to be positioned coaxially with respect to its storage cell. The 3-D model of an MPC-68 storage cell occupied with channeled BWR fuel is shown in Figure 4.4.4.

In the case of the PWR CSF, the porous medium extends to the entire cross-section of the storage cell. As described in [4.4.2], the CFD model for both the BWR and PWR case is prepared for the Design Basis fuel in comprehensive detail, which includes grid straps, BWR water rods and PWR guide and instrument tubes (assumed to be plugged for conservatism).

The axial flow through the rodded region under an impressed pressure differential between the two extremeties of the rodded space is computed in the deep laminar range for a set of Reynolds numbers. In this manner, a relationship between the pressure drop (ΔP) and axial flow is obtained for helium at reference conditions (7 atmospheres pressure and 450°F temperature) and the flow resistance parameters for input in the porous media simulation in the system 3-D model are obtained. This 3-D pressure vs. flow characterization of the rodded region by the porous media does not utilize any empirical loss correlations and forms an essential building block for the 3-D CFD model of the system.

c. Every MPC fuel storage cell is assumed to be occupied by design basis PWR or BWR fuel assemblies specified in Chapter 2 (Table 2.1.5). The in-plane thermal conductivity of the design basis fuel assemblies are obtained using ANSYS finite element models of an array of fuel rods enclosed by a square box. Radiation heat transfer from solid surfaces (cladding and box walls) are enabled in these models. Using these models the effective conduction-radiation conductivities are obtained and reported in Table 4.4.2. For heat transfer in the axial direction an area weighted mean of cladding and helium conductivities are computed (see Table 4.4.2). Axial conduction heat transfer in the fuel pellets and radiation heat dissipation in the axial direction are conservatively ignored. Thus, the thermal conductivity of the rodded region, like the porous media simulation for helium flow, is represented by a 3-D continuum having effective planar and axial conductivities.

d. The internals of the MPC, including the basket cross section, bottom mouse holes, top plenum, and circumferentially irregular downcomer are modeled explicitly. For simplicity, the mouse holes are modeled as rectangular openings with understated flow area.

e. The inlet and outlet vents in the HI-STORM overpack are modeled explicitly to incorporate any effects of non-axisymmetry of inlet air passages on the system's thermal performance.

f. The air flow in the HI-STORM/MPC annulus is simulated by a k-ω turbulence model with the transitional option enabled.

The 3-D model described above is illustrated in the cross section for the MPC-68 in Figure 4.4.3. A closeup of the fuel cell spaces which explicitly include the channel-to-cell gap in the 3-D model is shown in Figure 4.4.4. The principal 3-D modeling conservatisms are listed below:

1) The storage cell spaces are loaded with design basis fuel having the highest axial flow resistance (See Table 2.1.5).
2) Each storage cell is generating heat at it's limiting value under uniform or regionalized storage.
3) Axial dissipation of heat by the fuel pellets is neglected.
4) Axial dissipation of heat by radiation in the fuel bundle is neglected.
5) The fuel assembly channel length for BWR fuel is overstated.
6) The most severe environmental factors for long-term normal storage - ambient temperature of 80°F and 10CFR71 insolation levels - were coincidentally imposed on the system.

7) The absorbtivity of the external surfaces of the HI-STORM is conservatively assumed to be unity.
8) To understate MPC internal convection heat transfer, the helium pressure is understated.
9) No credit is taken for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports.
10) Heat dissipation by fuel basket peripheral supports is neglected.
11) Fuel basket and MPC shell emissivities are understated (see Table 4.2.4).
12) The k-ω model used for simulating the HI-STORM annulus flow yields uniformly conservative results [4.1.6].

The effect of crud resistance on fuel cladding surfaces has been evaluated and found to be negligible. The evaluation assumes a thick crud layer (130 μm) with a bounding low conductivity (conductivity of helium). The crud resistance increases the clad temperature by a very small amount (~0.1°F). Accordingly this effect is neglected in the thermal evaluations.

As we shall demonstrate in this chapter by comparing the results between axisymmetric and 3-D models, the former predicts higher fuel cladding temperatures. Having established the conservatism of the axisymmetric model the production runs are carried out using the axisymmetric model with the confidence that the reported results have an additional embedded conservatism in them.

4.4.1.2 Benchmarking the Porous Media Model for Peak Cladding Temperature

In the 3-D model described above, the rodded region inside the channel of BWR fuel bundles is replaced by a porous medium whose resistance is set so that the axial flow of helium under an impressed pressure difference between the two ends is equal to the flow in a GE-12/14 fuel bundle. However, while this establishes overall hydraulic resistance equivalence between the porous medium and GE-12/14 fuel, it does not address the effect of non-uniformity in the axial flow resistance in the actual fuel due to presence of bypass flows through the water rods (which are larger in diameter than fuel rods). Therefore, it is necessary to verify that, under the buoyancy driven helium flow, the water rod flow will not render the porous media clad temperature predictions in the non-conservative direction. For this purpose a 3-D CFD model of the BWR fuel in [4.1.6] was used to analyze the heat transfer problem.

A planar view of the GE-12/14 model is shown in Figure 4.4.8. Although not shown in the figure, the model includes eight 0.02 inch thick zircaloy grid straps in the rodded region. To simulate natural convection cooling of the fuel rods the channeled fuel bundle is assumed to be immersed in a large body of helium in a vertical orientation. In this orientation helium is propelled up the fuel bundle from the bottom by buoyancy forces created by the heating of helium, flows up through the fuel rods and water rod spaces and leaves from the top. The essential boundary conditions for the 3-D model are presented below:

Baseline Scenario
Helium Pressure - 7 atm
Sink temperature – 450°F

Fuel rods heat generation (q_o) – 0.6 kW
Channel outside heat transfer coefficient (h) – 1 Btu/ft^2-hr-°F

The same problem described above is also solved using a model that replaces the channel cross-section by a porous medium with the equivalent hydraulic parameters [4.4.2]. In this model the fuel rods heat generation is applied as a homogeneous volumetric heat source. The porous media thermal solution is obtained using the same boundary conditions (viz. pressure, temperature and channel outside heat transfer coefficient).

In addition to the baseline scenario defined above, sensitivity to assumed values of h and q_o are also examined. The table below provides a comparison of the peak cladding temperature from the 3-D and the porous media solutions for the baseline scenario and three sensitivity cases.

Scenario Description	q_o (kW)	h (Btu/ft^2-hr-°F)	Peak Cladding Temperature, °F	
			3-D Solution	Porous Media Solution
Baseline	0.6	1	550	559
½h	0.6	0.5	622	635
2h	0.6	2.0	512	518
High q_o	1	1	613	628

The value of h (heat transfer coefficient between the enveloping helium (heat sink) and the outside surface of the channel is in the range for channeled fuel cooled by laminar flows observed in the 3-D thermal simulation of the MPC in HI-STORM's analysis. The sink temperature is also in the general range that will exist around the hottest cell in the design basis thermal analysis of MPC-68, although it should be noted that the selection of the sink temperature will have a second order effect on the maximum temperature elevation (above the sink temperature) caused by the heat generation in the SNF. Finally, the specific (per assembly) heat generation rate (0.6 kW) is slightly larger than the Design Basis heat emission rate under uniform storage. The last case (1 kW) bounds the maximum specific heat generation rate under regionalized storage.

As can be seen from the parametric runs summarized above, the porous media predicts the peak cladding temperature with a modest level of conservatism with respect to the fully articulated 3-D thermosiphon model of the BWR fuel in its "channel".

4.4.1.3 Axisymmetric Model

The axisymmetric model requires several simplifications. In the most important step, the planar section of the MPC is homogenized. With each storage cell replaced with an equivalent solid square, the MPC cross section consists of a metallic gridwork (basket cell walls with each square cell space containing a solid storage cell square of effective thermal conductivity, which is a function of temperature) circumscribed by a circular ring (MPC shell). There are four distinct materials in this section, namely the homogenized storage cell squares, the Alloy X structural materials in the MPC (including neutron absorber sheathing), neutron absorber and helium gas. Each of the four constituent materials in this section has a different conductivity. It is emphasized that the conductivity of the homogenized storage cells is a strong function of temperature.

In the axisymmetric model, the required simplification is carried out by replacing the thermally heterogeneous fuel basket section mentioned above by an equivalent conduction-only region, using a finite element procedure. Because the rate of transport of heat within the fuel basket is influenced by radiation, which is a temperature-dependent effect, the equivalent conductivity of the fuel basket region must also be computed as a function of temperature. Also, it is recognized that the MPC section consists of two discrete regions, namely, the basket region and the peripheral region. The peripheral region is the space between the peripheral storage cells and the MPC shell. This is a helium filled space surrounded by Alloy X plates. Accordingly, as illustrated in Figure 4.4.2 for MPC-68, the MPC cross section is replaced with two homogenized regions with temperature-dependent conductivities. In particular, the effective conductivity of the storage cells is subsumed into the equivalent conductivity of the basket cross section, as described before.

As illustrated in Figure 4.4.2 an MPC fuel basket is composed of a welded gridwork of Alloy X-neutron absorber-sheathing sandwich panels to form an array of square shaped cells on a square pitch. The basket cell spaces are occupied by irradiated SNF. The two principal components of a loaded fuel basket – sandwich panels and SNF – have unequal conduction properties in the planar and axial directions. The fuel basket thermal modeling duly recognizes these differences by characterizing the effective conductivities in the two (planar and axial) directions. For computing the planar fuel basket conductivity, ANSYS finite element models of the PWR and BWR fuel baskets are employed. The principal inputs to the models are the design basis fuel planar conductivities and the sandwich panel conductivities discussed previously. The results of fuel basket modeling are presented in Table 4.4.3.

The fuel basket axial conductivity is computed by an area weighted sum of cladding, helium, neutron absorber and Alloy X (box wall and sheathing) conductivities. This evaluation employs the design basis fuel and neglects fuel pellets axial conduction and axial dissipation of heat by radiation .

Finally, HI-STORM is simulated as a radially symmetric structure having annular vents at the bottom and top with a buoyancy-induced flow in the annular space surrounding the heat generating MPC cylinder. The annular vents in the HI-STORM model are porous media spaces having effective inlet and outlet ducts flow resistances.

Internal circulation of helium in the sealed MPC is modeled as flow in a porous media in the fuel basket region containing the SNF (including top and bottom plenums). The basket-to-MPC shell clearance space is modeled as a helium filled radial gap to include the downcomer flow in the thermal model. The downcomer region, as illustrated in Figure 4.4.2, consists of an azimuthally varying gap formed by the square-celled basket outline and the cylindrical MPC shell. At the locations of closest approach a differential expansion gap (see Subsection 4.4.6) is engineered to allow free thermal expansion of the basket. At the widest locations, the gaps are on the order of the storage cell opening (~6" for BWR and ~9" for PWR MPCs). It is heuristically evident that heat dissipation by conduction is maximum at the closest approach locations (low thermal resistance path) and that convective heat transfer is highest at the widest gap locations (large downcomer flow). In the axisymmetric thermal model, a radial gap that is large compared to the basket-to-shell clearance and small compared to the cell opening is used. As a relatively large gap penalizes heat dissipation by conduction and a small gap throttles convective flow, the use of a single gap in the FLUENT model understates both conduction and convection heat transfer in the downcomer region.

To summarize, a loaded MPC standing upright on the ISFSI pad in a HI-STORM overpack is replaced with a right circular cylinder with spatially varying temperature-dependent conductivity. Heat is generated within the basket space in this cylinder in the manner of the prescribed axial burnup distribution. In addition, heat is deposited from insolation on the external surface of the overpack. Under steady state conditions the total heat due to internal generation and insolation is dissipated from the outer cask surfaces by natural convection, annulus ventilation cooling and thermal radiation to the ambient environment.

The table below provides a summary of the principal simplifications to the 3-D model that results in the computationally efficient axisymmetric model (which is also the supporting legacy model for early HI-STORM CoCs).

Comparison of 3-D and 2-D Model Attributes		
	3-D Model	**2-D Model**
a.	The storage cell and cavity along with SNF is modeled as a porous medium for helium axial flow. In the case of BWR fuel, the porous medium idealization is limited to the fuel channel/SNF space; the annular space between the fuel channel and cell walls is explicitly modeled.	The storage cell/cavity space is simulated by an equivalent porous medium for both PWR and BWR SNF. This model is subsumed in the fuel basket model (see item (c) below.
b.	The composite cell walls of the fuel basket (Alloy X-neutron absorber-sheathing sandwich) are replaced by thermally equivalent orthotropic solid panels.	Same as the 3-D model
c.	The fuel basket panels in (b) above are explicitly modeled.	The basket structure along with the SNF is replaced by an equivalent cylindrical body of equivalent temperature-dependent conductivity.
d.	The MPC shell and baseplate modeled explicitly.	Same as the 3-D model
e.	The "mouseholes" in the storage cell walls for helium circulation modeled explicitly.	Equivalent openings for axisymmetric helium circulation are provided in the model.
f.	The downcomer region (prismatic annulus of irregular cross section) is modeled explicitly.	The downcomer annulus region is defined by the space between the basked cylinder in (c) and the MPC shell. The

		width of the downcomer is conservatively set to ensure that the helium circulation rate will approach the actual from below.
g.	The HI-STORM inlet and outlet ducts are explicitly modeled.	The inlet and outlet ducts are simulated by equivalent flow resistance axisymmetric inlet and outlet passages.

It is evident from the foregoing that the 2-D model involves considerable simplifications. Therefore, to ensure that its use in production runs will not produce unconservative results, a number of comparison 3-D runs using identical input data were carried out, as discussed later in this section.

4.4.2 Comparison of Axisymmetric and 3-D Thermal Models

To confirm that the axisymmetric thermal model described in Subsection 4.4.1.2 is conservative, 3-D thermal models of the HI-STORM System are constructed as discussed previously in this section for bounding MPCs (MPC-68 for BWR fuel and MPC-32 for PWR fuel). For direct comparisons fuel storage scenarios under uniform and regionalized fuel loading in a HI-STORM are run and thermal solutions using both (axisymmetric and 3-D models) obtained. The results are compared in Table 4.4.4. In all cases the results show that the axisymmetric model yields conservative results.

In light of the above results, it can be concluded that whenever thermal evaluations reported in this FSAR employ the axisymmetric models, an additional layer of conservatism in the reported temperatures and pressures is implicitly embedded.

4.4.3 Test Model

The 3-D CFD modeling of the HI-STORM system with the conservatisms noted in the foregoing provides a high level of assurance that the predictions of the peak cladding temperature will be conservative. Prior to the 3-D model, all analyses were performed using a simpler 2-D model which had been benchmarked [4.1.5] with full-scale cask test data [4.1.3], as well as with PNNL's COBRA-SFS modeling of the HI-STORM System. All licensing basis thermal evaluations of the HI-STORM system through CoC Revision 2 were performed exclusively using the axisymmetric model described above.

The 3-D model eliminates virtually all of the simplifications of the 2-D model and utilizes proven codes, namely, the FLUENT CFD code and the industry standard ANSYS modeling package, to foster maximum confidence. Further, as discussed throughout this chapter and specifically in Subsection 4.4.1, the analysis incorporates significant conservatisms so as to compute fuel cladding temperatures in a bounding manner. Furthermore, compliance with specified limits of operation is demonstrated with adequate margins. In addition, experimental verification of thermal performance in the form of data from numerous previously loaded HI-STORMs has been compared with the predictions of the FLUENT CFD solutions and reported to the USNRC. These evaluations have shown that the FLUENT solutions are conservative in all cases. In view of these considerations, additional experimental verification of the thermal design is therefore not necessary at this time.

4.4.4 Maximum and Minimum Temperatures

4.4.4.1 Maximum Temperatures

The axisymmetric model benchmarked in the previous subsection is used to determine temperature distributions under long-term normal storage conditions for an array of cases covering PWR and BWR fuel storage in uniform and regionalized loading configurations. For this purpose one bounding MPC design in each of the two fuel classes – MPC-68 for BWR and MPC-32 for PWR – are analyzed and results obtained and summarized in this subsection. For a bounding evaluation the MPCs are assumed to be emplaced in a limiting overpack (HI-STORM 100S Version B).

For storage of PWR fuel two MPC designs, namely the MPC-24/24E and MPC-32 are available. Employing 3-D HI-STORM models fuel storage in both MPC-24 and MPC-32 canisters are evaluated. From the 3-D runs the peak cladding temperatures obtained in the MPC-24 are lower than the MPC-32 results by a substantial amount (~50°F). Thus, the MPC-32 is the bounding PWR-style fuel basket design. Accordingly HI-STORM thermal evaluations for PWR fuel storage employ the MPC-32 design. Results reported for this design bound the MPC-24/24E designs with additional margins.

The HI-STORM 100S Version B is the limiting overpack by virtue of the inlet and outlet vents design. Compared to two other overpack designs (i.e., HI-STORM 100 and HI-STORM 100S), the HI-STORM 100S Version B has smaller inlet and outlet vents. Thus Version B vent airflow resistances are bounding. Also, the HI-STORM 100S Version B is the shortest of the overpacks. This reduces the chimney height which minimizes the driving head for air flow. Because the HI-STORM 100S Version B will have the least cooling air flow, it will yield bounding results.

A cross-reference of HI-STORM thermal analyses is provided in Table 4.4.5. Under regionalized loading, an array of runs covering a range of regionalized storage configurations specified in Chapter 2 (X=0.5 to X=3) are analyzed. The results are graphed in Figures 4.4.6 and 4.4.7 for PWR and BWR fuel storage respectively. Based on this array of runs the fuel storage condition corresponding to X = 0.5 is determined to be limiting for both PWR and BWR MPCs. Accordingly HI-STORM MPC and overpack temperatures are reported for this storage condition in Tables 4.4.6 and 4.4.7.

It should be noted that the axisymmetric FLUENT cask model incorporates the effective conductivity results of the fuel basket submodel, which in turn incorporates the effective conductivity results of the fuel assembly submodel. Therefore the FLUENT model reports the peak temperature in every part of the system. In a dry storage cask, the hottest components are the fuel assemblies. Thus, as the fuel assembly models include the fuel pellets, the FLUENT calculated peak temperatures are actually peak pellet centerline temperatures which bound the peak cladding temperatures with a margin.

The following observations can be derived by inspecting the temperature field obtained from the axisymmetric thermal models:

- The fuel cladding temperatures are below the regulatory limit (ISG-11 [4.1.4]) under all storage scenarios (uniform and regionalized) in all MPCs.
- The maximum temperature of the basket structural materials are within their design limits.
- The maximum temperature of the neutron absorbers are below their design limits.
- The maximum temperatures of the MPC pressure boundary materials are below their design limits.
- The maximum temperatures of concrete is within the guidance of the governing ACI Code (see Table 4.3.1).

The above observations lead us to conclude that the temperature field in the HI-STORM System with a loaded MPC containing heat emitting SNF complies with all regulatory temperature limits. In other words, the thermal environment in the HI-STORM System is in compliance with Chapter 2 Design Criteria.

4.4.4.2 Minimum Temperatures

In Table 2.2.2 of this report, the minimum ambient temperature condition for the HI-STORM storage overpack and MPC is specified to be -40°F. If, conservatively, a zero decay heat load with no solar input is applied to the stored fuel assemblies, then every component of the system at steady state would be at a temperature of -40°F. Low service temperature (-40°F) evaluation of the HI-STORM is provided in Chapter 3. All HI-STORM storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode at this minimum temperature condition.

4.4.5 Maximum Internal Pressure

4.4.5.1 MPC Helium Backfill Pressure

The quantity of helium emplaced in the MPC cavity shall be sufficient to produce an operating pressure of 7 atmospheres (absolute) during normal storage at reference conditions (See Table 4.0.1). Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain minimum helium backfill pressure (P_b) specified at a reference temperature (70°F). The minimum backfill pressure for each MPC type is provided in Table 4.4.11. A theoretical upper limit on the helium backfill pressure also exists and is defined by the design pressure of the MPC vessel (Table 2.2.1). The upper limit of P_b is also reported in Table 4.4.11. To bound the minimum and maximum backfill pressures listed in Table 4.4.11 with a margin, a helium backfill specification is set forth in Table 4.4.12.

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

i. By pressure measurement
ii. By measurement of helium backfill volume (in standard cubic feet)

The direct pressure measurement approach is more convenient if the FHD method of MPC drying is used. In this case, a certain quantity of helium is already in the MPC. Because the helium is mixed inside the MPC during the FHD operation, the temperature of the helium gas at the MPC's exit, along with the pressure provides a reliable means to compute the inventory of helium using pressure and temperature gages. A shortfall or excess of helium is adjusted by a calculated raising or lowering of the MPC pressure such that the reference MPC backfill pressure is within the P_b specifications.

When vacuum drying is used as the method for MPC drying, then it is more convenient to fill the MPC by introducing a known quantity of helium (in standard cubic feet) by measuring the quantity of helium introduced using a calibrated mass flow meter or other measuring apparatus. The required quantity of helium is computed by the product of net free volume and helium specific volume at the reference temperature (70°F) and a target pressure that lies in the mid-range of the P_b specifications.

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

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

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

Using commercially available mass flow totalizers or other appropriate measuring devices, an MPC cavity is filled with the computed quantity of helium.

4.4.5.2 MPC Pressure Calculations

The MPC is initially filled with dry helium after fuel loading and drying prior to installing the MPC closure ring. During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined using the ideal gas law.

Table 4.4.8 presents a summary of the minimum MPC free volumes determined for each MPC type (MPC-24, MPC-68, MPC-32, and MPC-24E). The MPC maximum gas pressure is computed for a postulated release of fission product gases from fuel rods into this free space. For these scenarios, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the ideal gas law. Based on fission gases release fractions (NUREG 1536 criteria [4.4.1]), rods' net free volume and initial fill gas pressure, maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in Table 4.4.9. The maximum computed gas pressures reported in Table 4.4.9 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Table 2.2.1.

Evaluation of Non-Fuel Hardware

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the PWR baskets influences the MPC internal pressure through two distinct effects. The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures as well as the temperature of the gas filling the space between fuel rods. The gas volume displaced by the mass of non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influence on the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STORM thermal analysis, the computed temperature field (with non-fuel hardware excluded) has been determined to provide a conservatively bounding temperature field for the PWR baskets (MPC-24, MPC-24E, and MPC-32). The MPC cavity free space is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included. This approach ensures conservative bounding pressures.

During in-core irradiation of BPRAs, neutron capture by the B-10 isotope in the neutron absorbing material produces helium. Two different forms of the neutron absorbing material are used in BPRAs: Borosilicate glass and B_4C in a refractory solid matrix (Al_2O_3). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin-walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operation conditions (2,300 psia and approximately 600°F coolant temperature). The B_4C- Al_2O_3 neutron absorber material is principally used in B&W and CE fuel BPRA designs. The relatively low temperature of the poison material in BPRA rods (relative to fuel pellets) favor the entrapment of helium atoms in the solid matrix.

Several BPRA designs are used in PWR fuel that differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and W-15x15) has used 6, 12, 16, and 20 rods per assembly BPRAs and the later (W-17x17) fuel uses up to 24 rods per BPRA. The BPRA rods in the older fuel are much larger than the later fuel and, therefore, the B-10 isotope inventory in the 20-rod BPRAs bounds the newer W-17x17 fuel. Based on bounding BPRA rods internal pressure, a large hypothetical quantity of helium (7.2 g-moles/BPRA) is assumed to be available for release into the MPC cavity from each fuel assembly in the PWR baskets. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.9.

4.4.6 Engineered Clearances to Eliminate Thermal Interferences

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal stresses in load bearing members, the HI-STORM System is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansion calculations are performed to demonstrate that engineered gaps in the HI-STORM System are adequate to accommodate thermal expansion of the fuel basket and MPC.

The HI-STORM System is engineered with gaps for the fuel basket and MPC to expand thermally without restraint of free end expansion. Differential thermal expansion of the following gaps are evaluated:

a. Fuel Basket-to-MPC Radial Gap
b. Fuel Basket to MPC Axial Gap
c. MPC-to-Overpack Radial Gap
d. MPC-to-Overpack Axial Gap

To demonstrate that the fuel basket and MPC are free to expand without restraint, it is required to show that differential thermal expansion from fuel heatup is less than the as-built gaps that exist in the HI-STORM System. For this purpose a suitably bounding temperature profile (T(r)) for the fuel basket is established in Figure 4.4.5 wherein the center temperature (TC) is set at the limit (752°F) for fuel cladding (conservatively bounding assumption) and the basket periphery (TP) conservatively postulated at an upperbound of 600°F (see Table 4.4.6 for the maximum fuel and basket periphery temperatures). To maximize the fuel basket differential thermal expansion, the basket periphery-to-MPC shell temperature difference is conservatively maximized ($\Delta T = 175°F$). From the bounding temperature profile T(r) and ΔT, the mean fuel basket temperature (T1) and MPC shell temperature (T2) are computed as follows:

$$T1 = \frac{\int_0^1 rT(r)dr}{\int_0^1 rdr} = 676°F$$

$$T2 = TP - \Delta T = 425°F$$

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

$$Y1 = R \times \left[A1 \times (T1 - To) - A2 \times (T2 - To)\right]$$

where:

R = Basket radius (conservatively assumed to be the MPC radius)
A1, A2 = Coefficients of thermal expansion for fuel basket and MPC shell at T1 and T2 respectively for Alloy X (Chapter 1 and Table 3.3.1)

For computing the relative axial growth of the fuel basket in the MPC, bounding temperatures for the fuel basket (TC) and MPC shell temperature T2 utilized above are adopted. The differential expansion is computed by a formula similar to the one for radial growth after replacing R with basket height (H), which is conservatively assumed to be that of the MPC cavity.

For computing the radial and axial MPC-to-overpack differential expansions, the MPC shell is postulated at its design temperature (Chapter 2, Table 2.2.3) and thermal expansion of the overpack is ignored. Even with the conservative computation of the differential expansions in the manner of the foregoing, it is evident from the data compiled in Table 4.4.10 that the differential expansions are a fraction of their respective gaps.

4.4.7 Evaluation of System Performance for Normal Conditions of Storage

The HI-STORM System thermal analysis is based on a detailed and complete heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and overpack. The thermal model incorporates many conservative features that render the results for long-term storage to be extremely conservative:

Temperature distribution results obtained from this highly conservative thermal model show that the maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be much greater due to the many conservative assumptions incorporated in the analysis. The long-term impact of decay heat induced temperature levels on the HI-STORM System structural and neutron shielding materials is considered to be negligible. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep-induced degradation. Furthermore, stresses induced due to imposed temperature gradients are within Code limits (See Structural Evaluation Chapter 3). Therefore, it is concluded that the HI-STORM System thermal design is in compliance with 10CFR72 requirements.

Table 4.4.1

EFFECTIVE CONDUCTIVITY OF THE COMPOSITE FUEL BASKET WALLS
(Btu/hr-ft-°F)

	MPC-32		MPC-24/MPC-24E*		MPC-68	
Temperature (°F)	Thru-Thickness Direction	Parallel Plates Direction	Thru-Thickness Direction	Parallel Plates Direction	Thru-Thickness Direction	Parallel Plates Direction
200	6.000	14.65	5.676 4.800**	13.85 11.17**	5.544	12.06
450	7.260	16.12	6.864 5.808**	15.32 12.54**	6.708	13.45
700	8.316	17.20	7.884 6.672**	16.44 13.62**	7.680	14.52

* Lowerbound values reported.
** Effective conductivities of basket peripheral panels.

Table 4.4.2

LIMITING EFFECTIVE CONDUCTIVITIES OF THE RODDED REGION
(Btu/hr-ft-°F)

	PWR Fuel		BWR FUEL	
Temperature (°F)	Planar	Axial	Planar	Axial
200	0.257	0.753	0.282	0.897
450	0.406	0.833	0.425	0.988
700	0.604	0.934	0.606	1.104

Table 4.4.3

MPC BASKET EFFECTIVE THERMAL CONDUCTIVITIES
(Btu/hr-ft-°F)

	MPC-32		MPC-24/MPC-24E*		MPC-68	
Temperature (°F)	Planar	Axial**	Planar	Axial**	Planar	Axial**
200	1.061	1.773	1.143	2.427	1.073	2.186
450	1.324	1.933	1.554	2.666	1.307	2.379
700	1.603	2.079	2.047	2.870	1.550	2.548

* Lowerbound values reported.
** Conductivities are conservatively understated.

Table 4.4.4

COMPARISON OF 3-D AND AXISYMMETRIC MODEL SOLUTIONS

Regionalized Loading Parameter (X)	Peak Clad Temperature, °F		
	Axisymmetric Solution	3-D Solution	Relative Conservatism in the Axisymmetric Model, °F
MPC-68			
0.5	728	688	40
1	719	683	36
2	708	670	38
3	700	671	29
MPC-32			
0.5	720	710	10
1	715	706	9
2	716	704	12
3	713	702	11

Table 4.4.5

MATRIX OF HI-STORM SYSTEM THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Long Term Normal	Ambient	SS	N_T, Q_D, ST, SC, I_O	4.4.4
2	Off-Normal Environment	Ambient	SS(B)	O_T, Q_D, ST, SC, I_O	4.6.1
3	Extreme Environment	Ambient	SS(B)	E_T, Q_D, ST, SC, I_O	4.6.2
4	Partial Ducts Blockage	Ambient	SS(B)	N_T, Q_D, ST, SC, $I_{1/2}$	4.6.1
5	All Inlets Ducts Blocked	Overpack	TA	N_T, Q_D, ST, SC, I_C	4.6.2
6	Fire Accident	Overpack	TA	Q_D, F	4.6.2
7	Burial Under Debris	Overpack	AH	Q_D	4.6.2

Legend:

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

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

Table 4.4.6

MAXIMUM MPC TEMPERATURES FOR LONG-TERM NORMAL STORAGE CONDITION*

Component	Temperature, °F	
	MPC-32	MPC-68
Fuel Cladding	720	728
MPC Basket	605	711
Basket Periphery	563	580
MPC Outer Shell	480	490

Table 4.4.7

BOUNDING HI-STORM OVERPACK TEMPERATURES FOR LONG-TERM NORMAL STORAGE†

Component	Local Section Temperature‡, °F
Inner shell	356
Outer shell	159
Lid bottom plate	389
Lid top plate	213
Overpack Body Concrete	257
Overpack Lid Concrete	279
Air outlet†††	204

* The temperatures reported in this table are below the design temperatures specified in Chapter 2, Table 2.2.3.

† The temperatures reported in this table are below the design temperatures specified in Chapter 2, Table 2.2.3.

‡ Section temperature is defined as the through-thickness average temperature.

††† Reported herein for the option of temperature measurement surveillance of outlet ducts air temperature as set forth in the Technical Specifications.

Table 4.4.8

SUMMARY OF MPC-24 FREE VOLUME CALCULATIONS

Item	Volume (MPC-24) [ft^3]	Volume (MPC-24E) [ft^3]	Volume (MPC-32) [ft^3]	Volume (MPC-68) [ft^3]
Cavity Volume	367.9	367.9	367.9	367.3
Basket Metal Volume	44.3	51.4	24.9	34.8
Bounding Fuel Assemblies Volume	78.8	78.8	105.0	93.0
Basket Supports and Fuel Spacers Volume	6.1	6.1	9.0	11.3
Net Free Volume*	238.7 (6,759 liters)	231.6 (6,558 liters)	229 (6,484 liters)	228.2 (6,462 liters)

* Net free volumes are obtained by subtracting basket, fuel, supports and spacers metal volume from cavity volume. The free volumes used for MPC internal pressure calculations are conservatively understated.

Table 4.4.9

SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE*

Condition	MPC-24*** (psig)	MPC-24E*** (psig)	MPC-32 (psig)	MPC-68 (psig)
Initial backfill** (at 70°F)	47.5	47.5	47.5	47.5
Normal:				
intact rods	98.6	98.6	98.6	99.2
1% rods rupture	99.6	99.3	99.4	99.6
Off-Normal (10% rods rupture)	108.5	105.8	106.0	103.8
Accident (100% rods rupture)	197.4	169.9	172.0	145.1

* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

** Conservatively assumed at the Tech. Spec. maximum value (See Table 4.4.12).

*** Pressure calculations use the bounding MPC-32 temperature field.

Table 4.4.10

SUMMARY OF HI-STORM DIFFERENTIAL THERMAL EXPANSIONS

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

Table 4.4.11

THEORETICAL LIMITS* OF MPC HELIUM BACKFILL PRESSURE**

MPC	Minimum Backfill Pressure (psig)	Maximum Backfill Pressure (psig)
MPC-32/24/24E	43.4	48.3
MPC-68	43.1	48.0

* The helium backfill pressures are set forth in the Technical Specifications with a margin (See Table 4.4.12).

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

Table 4.4.12

MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS

Item	Specification
Minimum Pressure	44.0 psig @ 70°F Reference Temperature
Maximum Pressure	47.5 psig @ 70°F Reference Temperature

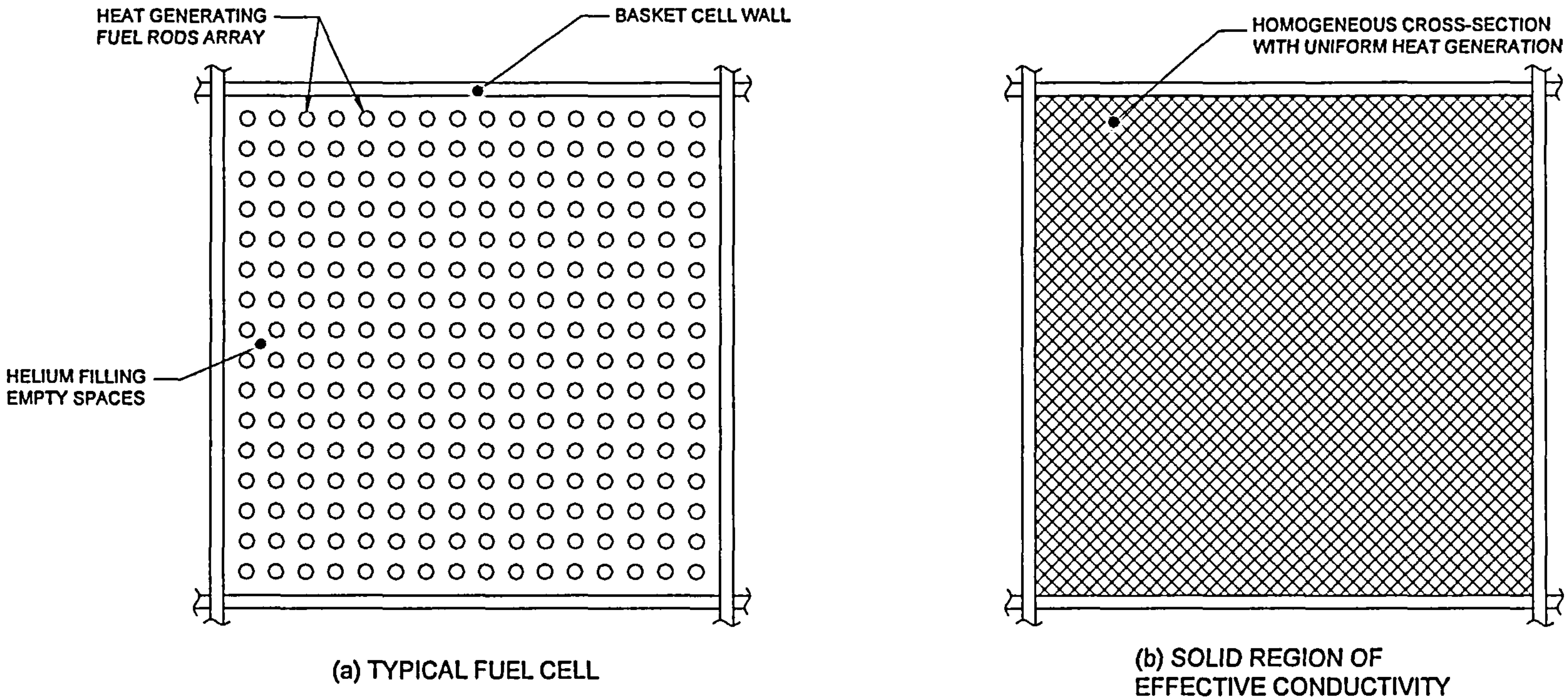

FIGURE 4.4.1: HOMOGENIZATION OF THE STORAGE CELL CROSS-SECTION

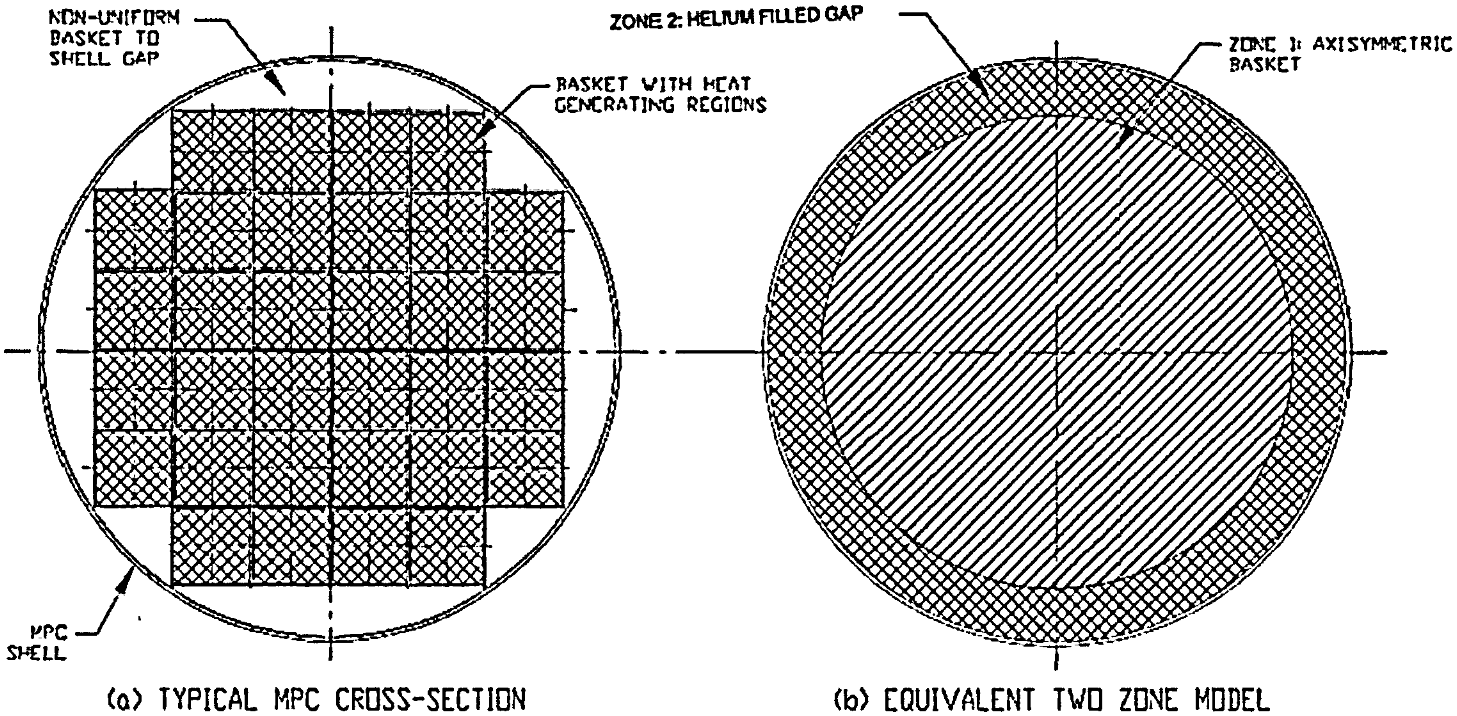

FIGURE 4.4.2: MPC CROSS-SECTION REPLACED WITH AN EQUIVALENT TWO ZONE AXISYMMETRIC BODY

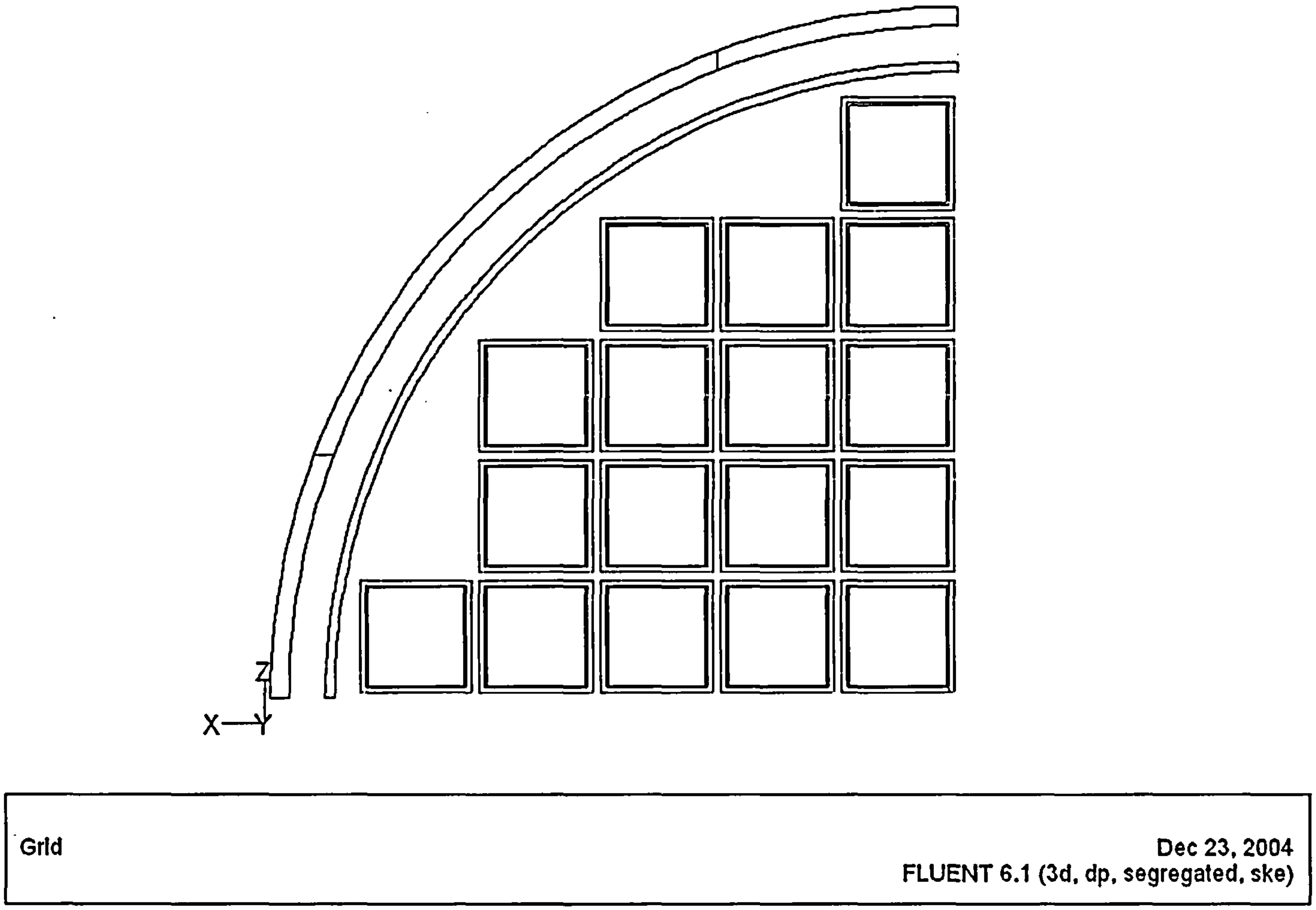

FIGURE 4.4.3: PLANAR VIEW OF HI-STORM MPC-68 QUARTER SYMMETRIC 3-D MODEL

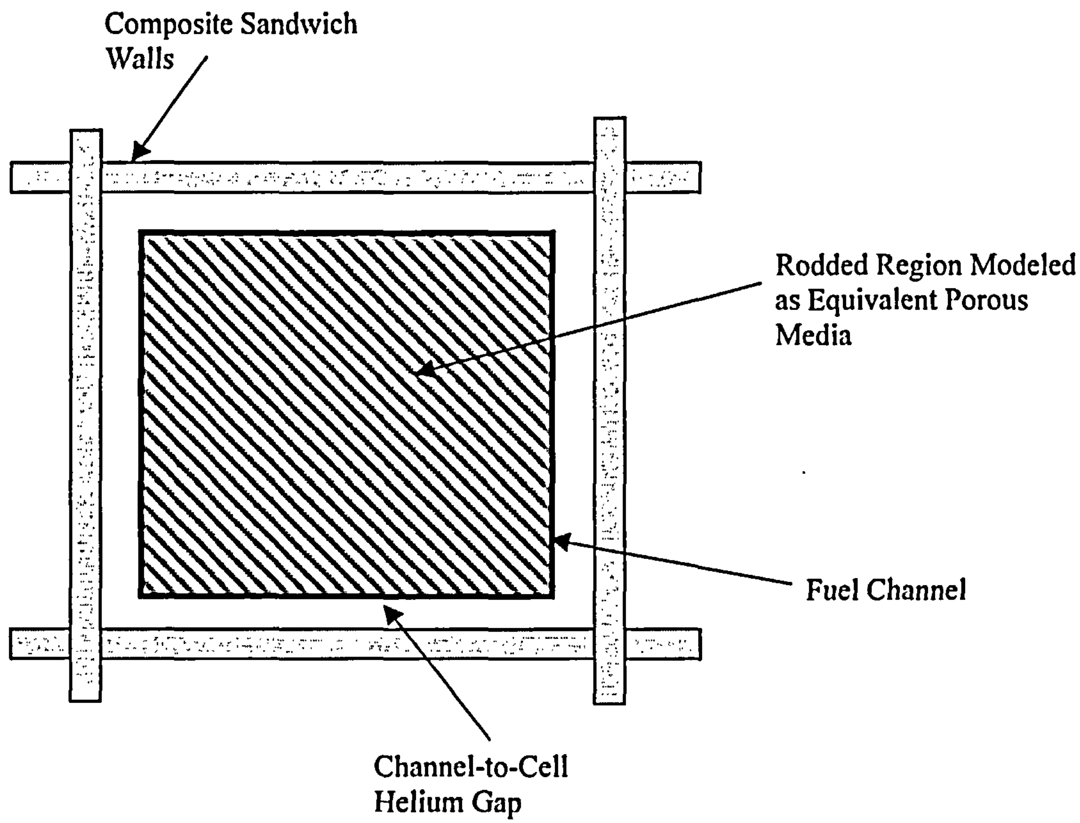

FIGURE 4.4.4: CLOSEUP VIEW OF THE MPC-68 CHANNELED FUEL CELL SPACES

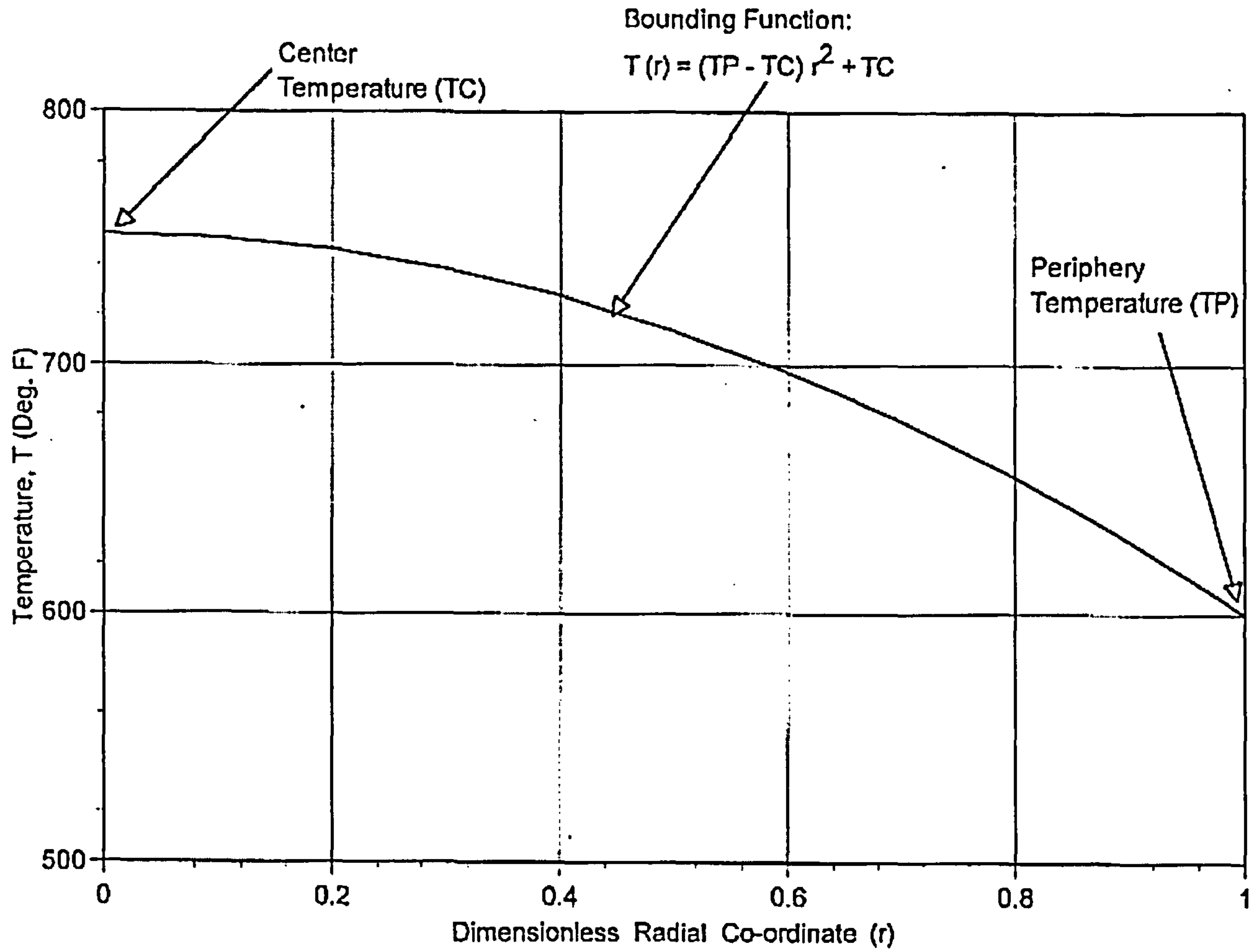

FIGURE 4.4.5: BOUNDING BASKET TEMPERATURE PROFILE FOR DIFFERENTIAL EXPANSION

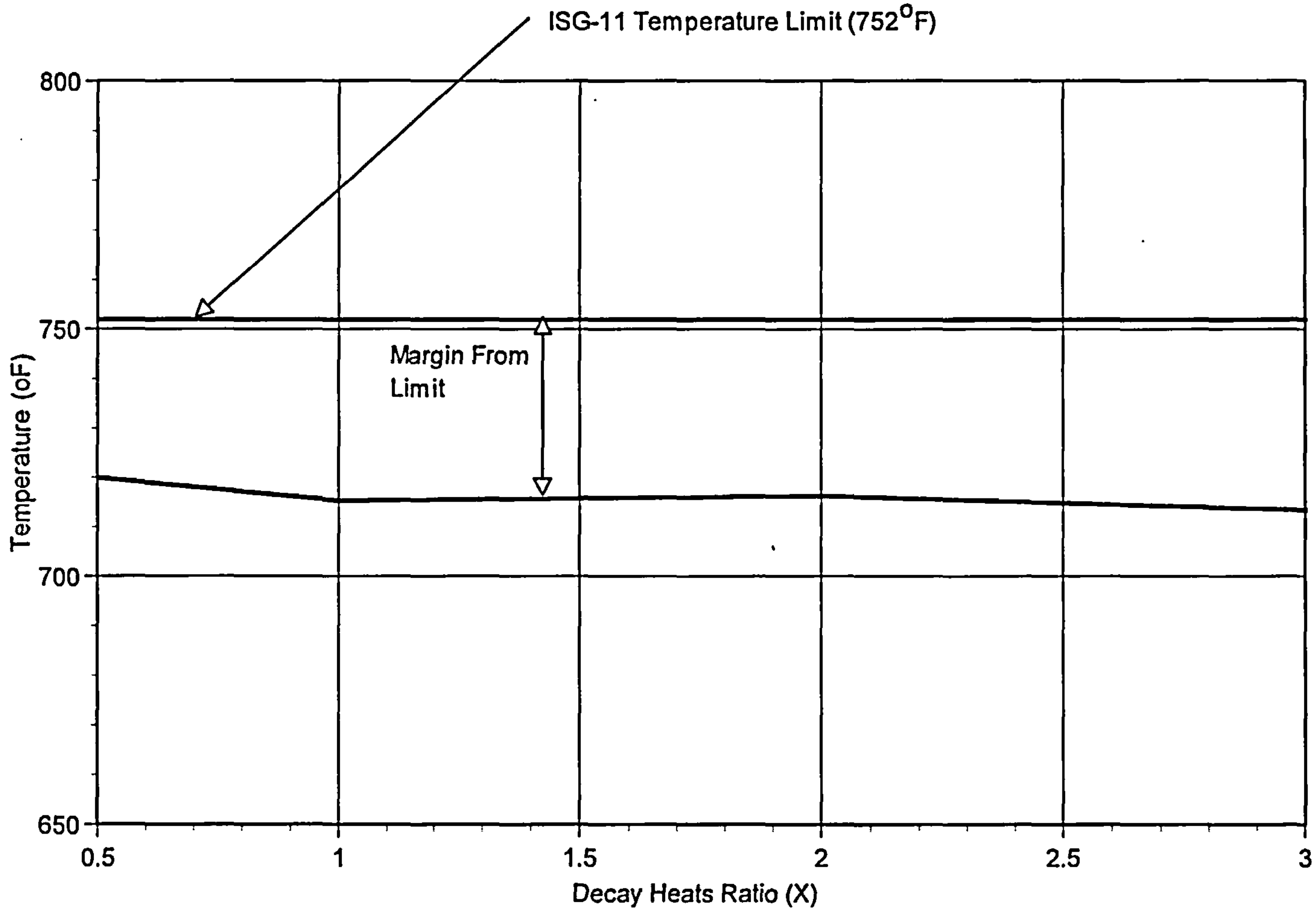

FIGURE 4.4.6: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC 32)

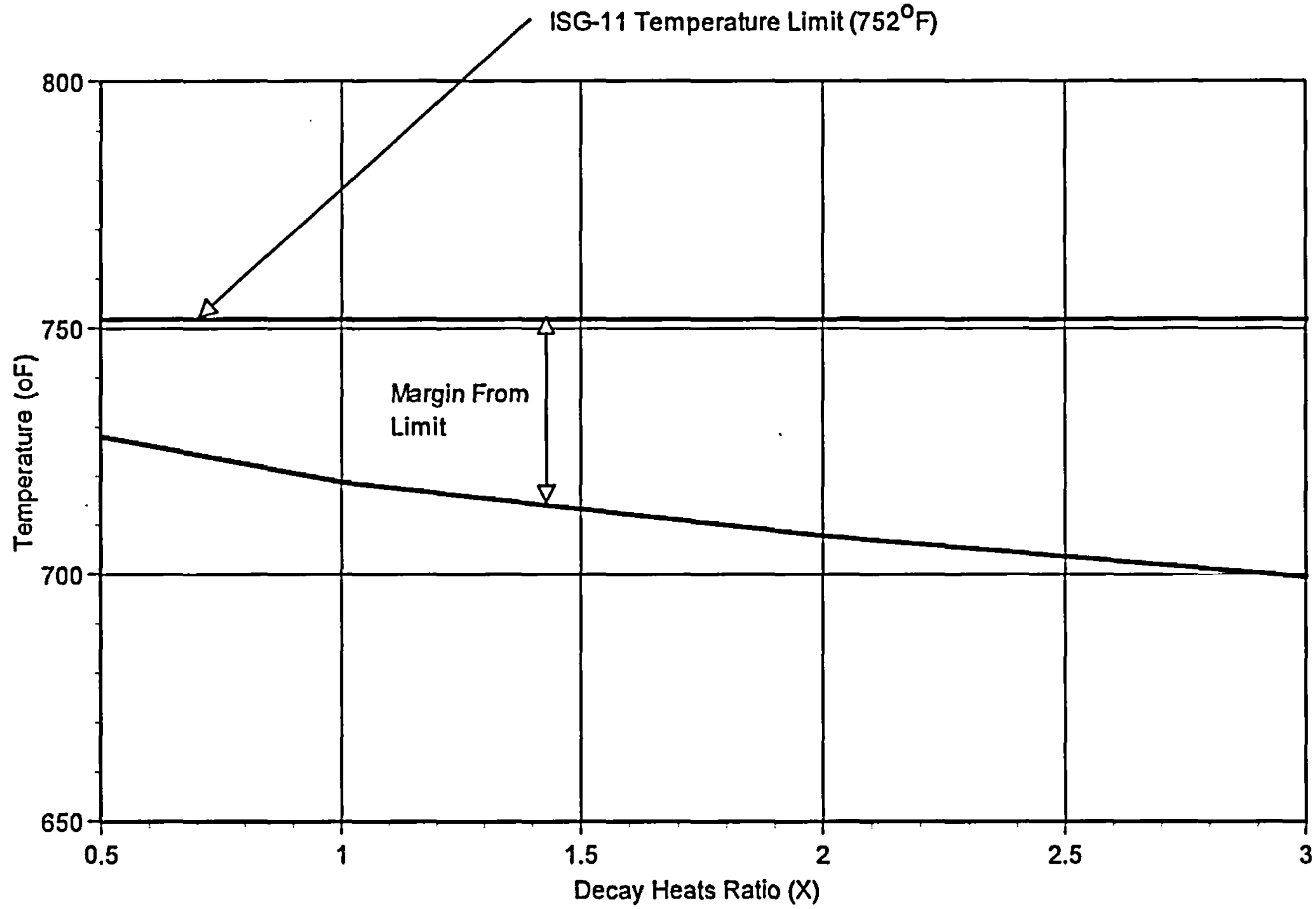

FIGURE 4.4.7: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-68)

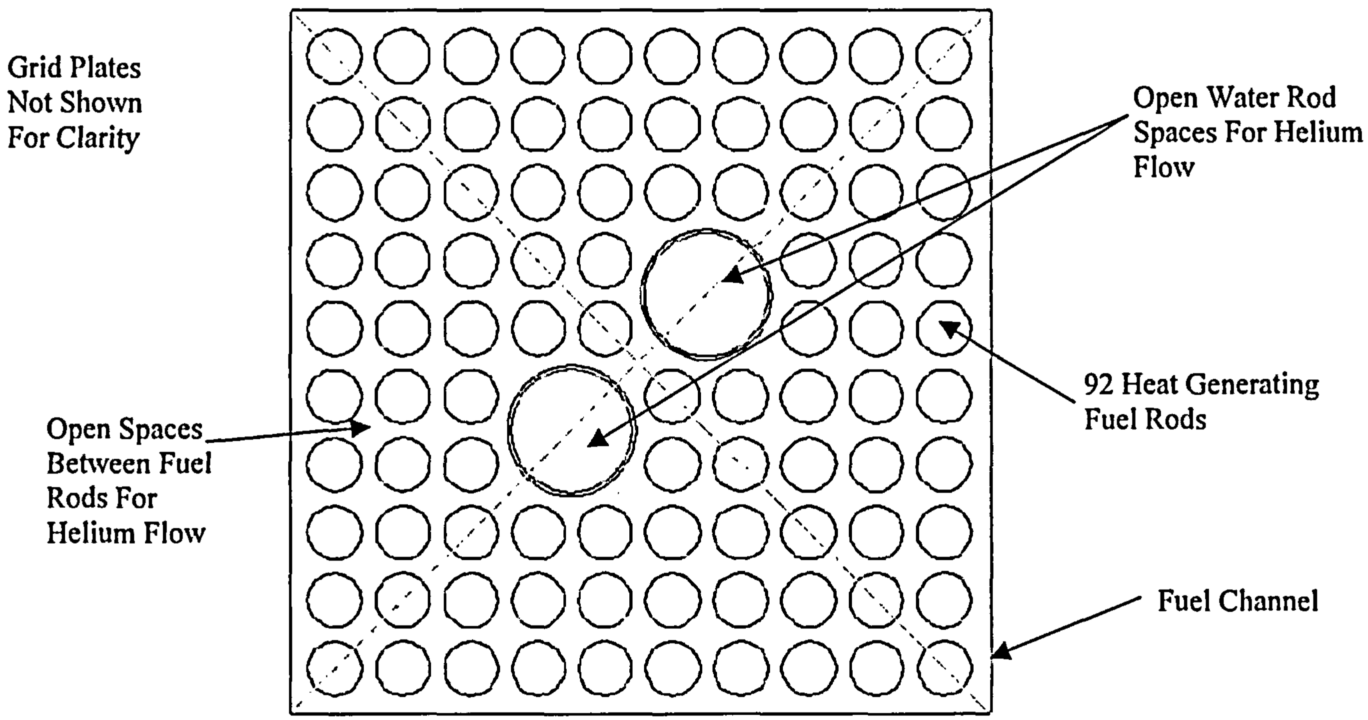

FIGURE 4.4.8: 3-D GE-12 FUEL ASSEMBLY THERMAL MODEL (PLANAR VIEW)

4.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

Prior to placement in a HI-STORM overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transported to the HI-STORM module. In the unlikely event that the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM overpacks or between a HI-STAR transport overpack and a HI-STORM storage overpack must be carried out in an assuredly safe manner. All of the above operations, henceforth referred to as "short term operations", are short duration events that would likely occur no more than once or twice for an individual MPC.

The device central to all of the above operations is the HI-TRAC transfer cask that, as stated in Chapter 1, is available in two anatomically similar weight ratings (100- and 125-ton). The HI-TRAC transfer cask is a short-term host for the MPC; therefore it is necessary to establish that, during all thermally challenging operation events involving either the 100-ton or 125-ton HI-TRAC, the permissible temperature limits presented in Section 4.3 are not exceeded. The following discrete thermal scenarios, all of short duration, involving the HI-TRAC transfer cask have been identified as warranting thermal analysis.

i. Post-Loading Wet Transfer Operations
ii. MPC Cavity Drying
iii. Normal Onsite Transport in a Vertical Orientation
iv. MPC Cooldown and Reflood for Unloading Operations

Onsite transport of the MPC occurs with the HI-TRAC in the vertical orientation, which preserves the thermosiphon action within the MPC. To avoid excessive temperatures, transport with the HI-TRAC in the horizontal condition is generally not permitted. However, it is recognized that an occasional downending of a HI-TRAC may become necessary to clear an obstruction such as a low egress bay door opening. In such a case the operational imperative for HI-TRAC downending must be ascertained and the permissible duration of horizontal configuration must be established on a site-specific basis and compliance with the thermal limits of ISG-11 [4.1.4] must be demonstrated as a part of the site-specific safety evaluation.

To ensure a bounding evaluation for the array of fuel storage configurations permitted in Section 2.1, a limiting storage condition is evaluated in this section. The limiting storage condition* is determined in Section 4.4.4 to be MPC-68 with regionalization parameter X = 0.5.

The fuel handling operations listed above place a certain level of constraint on the dissipation of heat from the MPC relative to the normal storage condition. Consequently, for some scenarios, it is necessary to provide additional cooling when certain threshold heat loads are exceeded. For

* Limiting fuel storage condition is defined as the fuel loading scenario that yields the highest computed fuel temperatures. From the array of analyses presented on Section 4.4, it is found that the highest clad temperatures co-incidentally occur with the highest permitted MPC heat load (i.e. for X = 0.5). Therefore the limiting scenario also yields the highest confinement boundary and overpack temperatures.

such situations, the Supplemental Cooling System (SCS) is required to provide additional cooling during short term operations. The SCS is required by the CoC for any MPC carrying one or more high burnup fuel assemblies when the MPC heat load is in excess of a conservatively postulated threshold ($Q^* = 25$ kW). For compliance with this specification, the MPC heat load is computed based on the discussion provided in Section 2.1 for uniform loading. In accordance with Section 2.1, the MPC heat load (Q) is computed by multiplying the number of fuel storage cells by the decay heat of the most emissive assembly loaded for storage. The SCS is required when Q exceeds Q^*. The specific design of an SCS must accord with site-specific needs and resources, including the availability of plant utilities. However, a set of specifications to ensure that the performance objectives of the SCS are satisfied by plant-specific designs are set forth in Appendix 2.C.

The short term operations listed above are described and evaluated in the following subsections. A map of HI-TRAC thermal evaluations is provided in Table 4.5.1.

4.5.1 HI-TRAC Thermal Model

The HI-TRAC transfer cask is used to load and unload the HI-STORM concrete storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI. Section views of the HI-TRAC have been presented in Chapter 1. Within a loaded HI-TRAC, heat generated in the MPC is transported from the contained fuel assemblies to the MPC shell in the manner described in Section 4.4. From the outer surface of the MPC to the ambient air, heat is transported by a combination of conduction, thermal radiation and natural convection. The axi-symmetric thermal model used for modeling the MPC is described in Section 4.4 and adopted for evaluation of short-term operations in a HI-TRAC overpack. Thermal modeling of the HI-TRAC transfer overpack is provided in this subsection.

Two HI-TRAC transfer cask designs, namely, the 125-ton and the 100-ton versions, are developed for onsite handling and transport, as discussed in Chapter 1. The two designs are principally different in terms of lead thickness and the thickness of radial connectors in the water jacket region. The analytical model developed for HI-TRAC thermal characterization conservatively accounts for these differences by applying the higher shell thickness and thinner radial connectors' thickness to the model. In this manner, the HI-TRAC overpack resistance to heat transfer is overestimated, resulting in higher predicted MPC internals and fuel cladding temperature levels.

The 100-ton and 125-ton HI-TRAC designs incorporate 2.5 inch and 4.5 inch annular spaces, respectively, formed between a 3/4-inch thick steel inner shell and a 1-inch thick steel outer shell. To ensure that lead forms a heat conduction continuum in the HI-TRAC body, solid lead bricks are not utilized in Holtec transfer casks. Rather, lead is poured in a molten state. The interior steel surfaces are cleaned, sandblasted and fluxed in preparation for the molten lead that will be poured in the annular cavity. The appropriate surface preparation technique is essential to ensure that molten lead sticks to the steel surfaces, which will form a metal to lead bond upon solidification. The molten lead is poured to fill the annular cavity. The molten lead in the immediate vicinity of the steel surfaces, upon cooling by the inner and outer shells, solidifies forming a melt-solid interface. The initial formation of a gap-free interfacial bond between the solidified lead and steel surfaces initiates a process of lead crystallization from the molten pool onto the solid surfaces. Static pressure from the column of molten lead further aids in retaining the solidified lead layer to the steel surfaces. The melt-solid interface growth occurs by freezing of successive layers of molten lead as the heat of fusion is dissipated by the solidified metal and steel structure enclosing it. This growth stops when all the molten lead is used up and the annulus is filled with a solid lead plug. The shop fabrication procedures, developed in conjunction with the manufacture of the HI-TRAC transfer casks contain detailed step-by-step instructions devised to eliminate the incidence of annular gaps in the lead space of the HI-TRAC. Accordingly the HI-TRAC transfer cask lead spaces are treated in the thermal models as continuous media.

Transport of heat within HI-TRAC occurs through multiple concentric layers of air, steel and shielding materials. From the surface of the enclosure shell heat is rejected to the atmosphere by natural convection and radiation.

A small diametral air gap exists between the outer surface of the MPC and the inner surface of the HI-TRAC overpack. Heat is transported across this gap by the parallel mechanisms of conduction and thermal radiation. Assuming that the MPC is centered and does not contact the transfer overpack walls conservatively minimizes heat transport across this gap. Thermal expansion would act to minimize this gap. At operating conditions, this gap would be quite small. For the purposes of evaluating heat transport across this gap, however, it is conservatively assumed that the gap is only reduced to one-half of its nominal value. Heat is transported through the cylindrical wall of the HI-TRAC transfer overpack by conduction through successive layers of steel, lead and steel. A water jacket, which provides neutron shielding for the HI-TRAC overpack, surrounds the cylindrical steel wall. The water jacket is essentially an array of carbon steel radial ribs with welded, connecting enclosure plates. Heat is dissipated by conduction and natural convection in the water cavities and by conduction in the radial ribs. Heat is passively rejected to the ambient from the outer surface of the HI-TRAC transfer overpack by natural convection and thermal radiation.

The HI-TRAC bottom is conservatively modeled as an insulated surface. The HI-TRAC top lid and sides are modeled as insolation heated surfaces cooled by convection and radiation. Insolation on exposed surfaces is conservatively based on 12-hour insolation inputs from 10CFR71 averaged on a 24-hour basis.

4.5.1.1 Effective Thermal Conductivity of Water Jacket

The HI-TRAC water jacket is composed of an array of radial ribs equispaced along the circumference of the HI-TRAC. Enclosure plates are welded to these ribs, creating an array of water compartments. Holes in the radial ribs connect all the individual compartments in the water jacket. The annular region between the HI-TRAC outer shell and the enclosure shell is an array of steel ribs and water spaces.

The effective radial thermal conductivity of this array of steel ribs and water spaces is determined by combining the heat transfer resistance of individual components (steel ribs and water spaces) in a parallel network. A bounding calculation is assured by using a minimum available metal thickness (product of number of radial ribs and rib thickness) for radial heat transfer.

The water in the jacket is free to move under the effects of buoyancy forces. The effect of this water motion on heat transfer is characterized by the Nusselt number (Nu), which can be defined as follows for a vertical enclosure [4.5.1]:

$$Nu = 0.046 \times Ra^{1/3}$$

where Ra is the Rayleigh number. For a conservatively determined Rayleigh number, based on the radial width of the water space, the Nusselt number for the water in the water jacket is approximately 79. This value is used as a multiplier on the thermal conductivity of water in the water jacket to reflect the effects of water motion on heat transfer in this region.

4.5.1.1.2 Heat Rejection from Overpack Exterior Surfaces

The following relationship is used for modeling heat loss from exposed cask surfaces:

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

where:

T_S = cask surface temperatures (°F)
T_A = ambient atmospheric temperature (°F)
q_s = surface heat flux ($Btu/ft^2 \times hr$)
ε = surface emissivity

The second term in this equation the Stefan-Boltzmann formula for thermal radiation from an exposed surface to ambient. The first term is the natural convection heat transfer correlation recommended by Jacob and Hawkins [4.2.9]. This correlation is appropriate for turbulent natural convection from vertical surfaces, such as the vertical overpack wall. Although the ambient air is conservatively assumed to be quiescent, the natural convection is nevertheless turbulent.

Turbulent natural convection correlations are suitable for use when the product of the Grashof and Prandtl (Gr×Pr) numbers exceeds 10^9. This product can be expressed as $L^3 \times \Delta T \times Z$, where L is the characteristic length, ΔT is the surface-to-ambient temperature difference, and Z is a function of the surface temperature. The characteristic length of a vertically oriented HI-TRAC is its height of approximately 17 feet. The value of Z, conservatively taken at a surface temperature of 340°F, is 2.6×10^5. Solving for the value of ΔT that satisfies the equivalence

$L^3 \times \Delta T \times Z = 10^9$ yields $\Delta T = 0.78°F$. The natural convection will be turbulent, therefore, provided the surface to air temperature difference is greater than or equal to 0.78°F.

4.5.1.3 Determination of Solar Heat Input

The thermal evaluations use the 10CFR71 specified 12-hour insolation as a 24-hour averaged heat flux on exposed HI-TRAC surfaces. This is appropriate, as the HI-TRAC cask possesses a considerable thermal inertia that precludes it from reaching steady state during a 12-hour insolation period.

4.5.2 Maximum Time Limit During Wet Transfer Operations

In accordance with NUREG-1536, water inside the MPC cavity during wet transfer operations is not permitted to boil. Consequently, uncontrolled pressures in the de-watering, purging, and recharging system that may result from two-phase conditions are completely avoided. This requirement is accomplished by imposing a limit on the maximum allowable time duration for fuel to be submerged in water after a loaded HI-TRAC cask is removed from the pool and prior to the start of vacuum drying operations.

Fuel loading operations are typically conducted with the HI-TRAC and it's contents (water filled MPC) submerged in pool water. Under these conditions, the HI-TRAC is essentially at the pool water temperature. When the HI-TRAC transfer cask and the loaded MPC under water-flooded conditions is removed from the pool, the water, fuel, MPC and HI-TRAC metal absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the HI-TRAC with time, starting from an initial (pool water) temperature. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC system. To enable a bounding heat-up rate determination, the following conservative assumptions are utilized:

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

ii. Design maximum decay heat input from the loaded fuel assemblies is assumed.

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

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

Table 4.5.2 summarizes the weights and thermal inertias of several components in the loaded HI-TRAC transfer cask. The rate of temperature rise of the HI-TRAC transfer cask and contents during an adiabatic heat-up is governed by the following equation:

$$\frac{dT}{dt} = \frac{Q}{C_h}$$

where:

- Q = conservatively bounding heat load (Btu/hr) [38 kW = 1.3×10^5 Btu/hr]
- C_h = thermal inertia of a loaded HI-TRAC (Btu/°F)
- T = temperature of the HI-TRAC cask (°F)
- t = time after HI-TRAC transfer cask is removed from the pool (hr)

A bounding heat-up rate for the HI-TRAC transfer cask contents is determined to be equal to 4.99°F/hr. From this adiabatic rate of temperature rise estimate, the maximum allowable time duration (t_{max}) for fuel to be submerged in water is determined as follows:

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

where:

- T_{boil} = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)
- $T_{initial}$ =initial HI-TRAC temperature when the transfer cask is removed from the pool

Table 4.5.3 provides a summary of t_{max} at several representative initial temperatures.

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

$$M_W = \frac{Q}{C_{pW}\,(T_{max} - T_{in})}$$

where:

- M_W = minimum water flow rate (lb/hr)
- C_{pw} = water heat capacity (Btu/lb-°F)
- T_{max} = maximum MPC cavity water mass temperature
- T_{in} = temperature of pool water supply to MPC

With the MPC cavity water temperature limited to 150°F, MPC inlet water maximum temperature equal to 125°F and at the design basis maximum heat load, the water flow rate is determined to be 5210 lb/hr (10.5 gpm).

4.5.3 MPC Temperatures During Moisture Removal Operations

4.5.3.1 Vacuum Drying Operation

The initial loading of SNF in the MPC requires that the water within the MPC be drained and replaced with helium. For MPCs containing moderate burnup fuel assemblies only, this operation may be carried out using the conventional vacuum drying approach. In this method, removal of the last traces of residual moisture from the MPC cavity is accomplished by evacuating the MPC for a short time after draining the MPC. Vacuum drying of MPCs containing any high burnup fuel assemblies is not permitted. High burnup fuel drying is performed by a forced flow helium drying process as described in Appendix 2.B.

Prior to the start of the MPC draining operation, both the HI-TRAC annulus and the MPC are full of water. The presence of water in the HI-TRAC annulus ensures adequate fuel cooling even under high vacuum (~1 torr) for extended durations. As the heat generating active fuel length is uncovered during MPC draining operation, the fuel and basket mass will undergo a gradual heat up from the initially cold conditions when the heated surfaces were submerged under water.

Vacuum drying is evaluated assuming the MPC space is filled with water vapor at a very low pressure (1 torr) and bounding steady state temperatures have reached. For allowing vacuum drying of MBF without time limit restrictions MPC dependent threshold heat load are specified below:

MPC-24/24E: 29 kW
MPC-32: 26 kW
MPC-68: 26 kW

For total decay heat loads up to threshold heat loads, vacuum drying of the MPC is permitted with the annular gap between the MPC and the HI-TRAC filled with water. The presence of water in this annular gap will maintain the MPC shell temperature approximately equal to the saturation temperature of the annulus water. In the vacuum drying thermal analysis a bounding MPC shell temperature (232°F) is conservatively assumed. Axisymmetric FLUENT thermal models of the PWR and BWR MPCs (MPC-24/24E, MPC-32 and MPC-68) are constructed, employing the following bounding assumptions:

i. Bounding steady-state condition.

ii. The MPC shell is postulated to be at a bounding maximum temperature of 232°F.

iii. The top surface of the MPC is cooled to the ambient

iv. The bottom surface of the MPC is insulated.

An axisymmetric FLUENT thermal model of the MPC is constructed for vacuum drying of moderate burnup fuel*. Each MPC is analyzed at its respective threshold heat load defined previously and fuel cladding temperatures below prescribed limit for MBF (Table 4.3.1) confirmed.

4.5.3.2 Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a closed loop dehumidification system consisting of a condenser, a demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. A vapor pressure of 3 torr or less is assured by verifying that the helium temperature exiting the demoisturizer is maintained at or below the psychrometric threshold of 21°F for a minimum of 30 minutes. See Appendix 2.B for detailed discussion of the design criteria and operation of the FHD system.

The FHD system provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHD system ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit for normal conditions of storage, which is well below the high burnup cladding temperature limit 752°F (400°C) for all combinations of SNF type, burnup, decay heat, and cooling time. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode of natural convection in long term storage), it is readily concluded that the peak fuel cladding temperature under the latter condition will be greater than that during the FHD operation phase. In the event that the FHD system malfunctions, the forced convection state will degenerate to natural convection, which corresponds to the conditions of normal onsite transport. As a result, the peak fuel cladding temperatures will approximate the values reached during normal onsite transport as described elsewhere in this chapter.

4.5.4 Maximum Temperatures Under Onsite Transport Conditions

An axisymmetric FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was constructed to evaluate temperature distributions for onsite transport. A bounding steady-state analysis of the HI-TRAC transfer cask has been performed for a limiting fuel storage configuration (X = 0.5, MPC-68). While the duration of onsite transport may be short enough to preclude the MPC and HI-TRAC from reaching steady-state, a steady-state analysis is conservative.

The maximum HI-TRAC onsite transport temperatures are reported in Table 4.5.4. The results satisfy the temperature limits for moderate burnup fuel (see Table 4.3.1). For high burnup fuel

* Vacuum drying of high burnup fuel is not permitted. MPCs containing one or more HBF assemblies shall be de-moisturized using the FHD drying method.

(HBF) the maximum computed fuel cladding temperature reported in Table 4.5.4 is greater than the temperature limit of 752°F for HBF. Consequently, it is necessary to utilize the SCS described at the beginning of this section and specified in Appendix 2.C during onsite transfer of an MPC with a greater than threshold heat load and containing one or more HBF assemblies. As stated earlier, the exact design and operation of the SCS is necessarily site-specific. The design is required to satisfy the design and operational requirements of Appendix 2.C to ensure compliance with ISG-11 [4.1.4] temperature limits.

4.5.5 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

NUREG-1536 requires an evaluation of cask cooldown and reflooding to support fuel unloading from a dry condition. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. For high heat load MPCs, the extremely rapid cooldown rates to which the hot MPC internals and the fuel cladding can be subjected during water injection may, however, result in high thermal stresses. Additionally, water injection may also result in some steam generation. To limit the fuel cladding from thermal strains from direct water quenching, the MPCs may be cooled using appropriate means prior to the introduction of water in the MPC cavity space.

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

Under a closed-loop forced helium circulation condition, the helium gas is cooled, via an external chiller. The chilled helium is then introduced into the MPC cavity from connections at the top of the MPC lid. The helium gas enters the MPC basket and moves through the fuel basket cells, removing heat from the fuel assemblies and MPC internals. The heated helium gas exits the MPC from the lid connection to the helium recirculation and cooling system. Because of the turbulation and mixing of the helium contents in the MPC cavity by the forced circulation, the MPC exiting temperature is a reliable measure of the thermal condition inside the MPC cavity. The objective of the cooldown system is to lower the bulk helium temperature in the MPC cavity to below the normal boiling temperature of water (212°F). For this purpose, the rate of helium circulation shall be sufficient to ensure that the helium exit gas temperature is below this threshold limit with a margin.

An example calculation for the required helium circulation rate is provided below to limit the helium temperature to 200°F. The calculation assumes no heat loss from the MPC boundaries and a conservatively bounding heat load (38 kW ($1.3x10^5$ Btu/hr)). Under these assumptions, the MPC helium is heated adiabatically by the MPC decay heat from a given inlet temperature (T_1) to a temperature (T_2). The required circulation rate to limit T_2 to 200°F is computed as follows:

$$m = \frac{Q_d}{C_p(T_2 - T_1)}$$

where:

Q_d = Design maximum decay heat load (Btu/hr)
m = Minimum helium circulation rate (lb/hr)
Cp = Heat capacity of helium (1.24 Btu/lb-°F (Table 4.2.5))
T_1 = Helium supply temperature (assumed 15°F in this example)

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

4.5.6 Maximum Internal Pressure

After fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling and on-site transfer operations in the HI-TRAC transfer cask, the gas temperature within the MPC rises to its maximum operating temperature as determined based on the thermal analysis methodology described previously. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined based on the ideal gas law. The maximum MPC internal pressure is determined using a bounding set of assumptions, namely:

a) Limiting fuel storage condition
b) Steady state maximum temperatures have reached
c) HI-TRAC in a 80°F quiescent ambient temperature
d) HI-TRAC annulus is filled with a stationary air column
e) Exposed surfaces of cask heated by insolation
f) MPC backfilled with helium to Technical Specification maximum level

Under the adverse set of conditions defined above the maximum MPC pressure is computed and compared with the short term (off-normal) pressure limit specified in Table 2.2.1. The computed result (See Table 4.5.4) meets the pressure limit with a margin.

4.5.7 Evaluation of HI-TRAC Performance for Short Term Operations

The HI-TRAC transfer cask thermal analysis is based on a detailed heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and HI-TRAC. The thermal model incorporates several conservative features, which are listed below:

i. A constant solar flux is imposed in the thermal model. A bounding solar absorbtivity of 1.0 is applied to all insolation surfaces.

ii. The MPC is considered to be concentrically aligned within the cask cavity. This is a worst-case scenario since any eccentricity will improve conductive heat transport in this region.

iii. No credit is considered for cooling of the HI-TRAC baseplate while in contact with a supporting surface. An insulated boundary condition is applied in the thermal model on the bottom baseplate face.

Temperature distribution results (Tables 4.5.4) obtained from this conservative thermal model show that the fuel cladding and cask component temperature limits are met with adequate margins for MBF. For HBF, supplemental cooling is specified to comply with the applicable temperature limits. Expected margins during HI-TRAC operations will be larger due to the many conservative assumptions incorporated in the analysis. Corresponding MPC internal pressure remains below the short-term condition design pressure. The maximum local neutron shield temperature is lower than design limits. Therefore, it is concluded that the HI-TRAC transfer cask thermal design is adequate to maintain fuel cladding integrity for short-term onsite handling and transfer operations.

The water in the water jacket of the HI-TRAC provides necessary neutron shielding. During normal handling and onsite transfer operations this shielding water is contained within the water jacket, which is designed for an elevated internal pressure. It is recalled that the water jacket is equipped with pressure relief valves set at 60 psig and 65 psig. This set pressure elevates the saturation pressure and temperature inside the water jacket, thereby precluding boiling in the water jacket under normal conditions. Under normal handling and onsite transfer operations, the bulk temperature inside the water jacket reported in Table 4.5.4 is less than the coincident saturation temperature at 60 psig (307°F), so the shielding water remains in its liquid state. The bulk temperature is determined via a conservative analysis, presented earlier, for a limiting fuel storage configuration.

During a hypothetical fire accident conditions these relief valves allow venting of steam to prevent overpressurizing the water jacket. In this manner, a portion of the fire heat flux input to the HI-TRAC outer surfaces is expended in vaporizing water in the water jacket, thereby mitigating the magnitude of the heat input to the MPC during a fire.

During vacuum drying operations, the annular gap between the MPC and the HI-TRAC is filled with water. The saturation temperature of the annulus water bounds the maximum temperatures of all HI-TRAC components, which are located radially outside the water-filled annulus. As previously stated (see Subsection 4.5.3) the maximum annulus water temperature is 232°F, so the HI-TRAC water jacket temperature will be less than the 307°F saturation temperature.

Table 4.5.1

MATRIX OF HI-TRAC TRANSFER THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Onsite Transport	Ambient	SS(B)	Q_D, ST, SC	4.5.4
3	Vacuum	HI-TRAC Annulus Water	SS(B)	Q_D	4.5.3
3	Wet Transfer Operation	Cavity Water and Cask Internals	AH	Q_D	4.5.2
4	Fuel Unloading	Helium Circulation	TA	Q_D	4.5.5
5	Fire Accident	Jacket Water, Cask Internals	TA	Q_D, F	4.6.2
6	Jacket Water Loss Accident	Ambient	SS(B)	O_T, Q_D, ST, SC	4.6.2

Legend:

Q_D - Design Basis Maximum Heat Load
ST - Insolation Heating (Top Surface)
SC - Insolation Heating (Curved Surfaces)
F - Fire Heating (1475°F)

SS(B) – Bounding Steady State
TA - Transient Analysis
AH - Adiabatic Heating

Table 4.5.2

HI-TRAC TRANSFER CASK LOWERBOUND
WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
Water Jacket	7,000	1.0	7,000
Lead	52,000	0.031	1,612
Carbon Steel	40,000	0.1	4,000
Alloy-X MPC (empty)	39,000	0.12	4,680
Fuel	40,000	0.056	2,240
MPC Cavity Water*	6,500	1.0	6,500
			26,032 (Total)

* Conservative lower bound water mass.

Table 4.5.3

MAXIMUM ALLOWABLE TIME FOR WET
TRANSFER OPERATIONS

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

Table 4.5.4

HI-TRAC ONSITE TRANSPORT MAXIMUM TEMPERATURES AND PRESSURES

Component	Temperature [°F]*
Fuel Cladding	777**
MPC Basket	764
Basket Periphery	601
MPC Outer Shell Surface	507
HI-TRAC Inner Shell	386
HI-TRAC Enclosure Shell	269
Water Jacket Bulk Water	252
Axial Neutron Shield	328
Pressure (psig)	
MPC	105

* The reported temperatures are below the HI-TRAC short-term temperature limits (Table 2.2.3).

** The reported temperature exceeds the allowable temperature limit for HBF fuel. The Supplemental Cooling System described in Appendix 2.C is required for greater than threshold heat load MPCs containing one or more HBF assemblies. (See threshold heat load discussion near the beginning of Section 4.5.)

4.6 OFF-NORMAL AND ACCIDENT EVENTS[1]

In accordance with NUREG 1536 the HI-STORM 100 System is evaluated for the effects of off-normal and accident events. The design basis off-normal and accident events are defined in Chapter 2. For each event, the cause of the event, means of detection, consequences, and corrective actions are discussed and evaluated in Chapter 11. To support the Chapter 11 evaluations, thermal analyses of limiting off-normal and accident events are provided in the following.

To ensure a bounding evaluation for the array of fuel storage configurations permitted in Section 2.1, a limiting storage condition is evaluated in this section. The limiting storage condition is previously determined in the Section 4.5 and adopted herein for all off-normal and accident evaluations.

4.6.1 Off-Normal Events

4.6.1.1 Off-Normal Pressure

This event is defined as a combination of (a) maximum helium backfill pressure (Table 4.4.12), (b) 10% fuel rods rupture, and (c) limiting fuel storage configuration. The principal objective of the analysis is to demonstrate that the MPC off-normal design pressure (Table 2.2.1) is not exceeded. The MPC off-normal pressures are reported in Table 4.4.9. The result[2] is confirmed to be below the off-normal design pressure (Table 2.2.1).

4.6.1.2 Off-Normal Environmental Temperature

This event is defined by a time averaged ambient temperature of 100°F for a 3-day period (Table 2.2.2). The consequences of this event are bounded by the "Off-Normal Pressure" event evaluated earlier in this subsection.

4.6.1.3 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with debris screens installed on the inlet and outlet openings. These screens ensure the air passages are protected from entry and blockage by foreign objects. As required by the design criteria presented in Chapter 2, it is postulated that two of the four air inlet ducts in the aboveground HI-STORM overpack are blocked. The resulting decrease in flow area increases the flow resistance of the inlet ducts. The effect of the increased flow resistance on fuel temperature is analyzed for the normal ambient temperature (Table 2.2.2) and a limiting fuel storage configuration. The computed temperatures are reported in Table 4.6.1 and the corresponding MPC internal pressure in Table 4.6.2. The results are confirmed to be below the temperature limits (Table 2.2.3) and pressure limit (Table 2.2.1) for off-normal conditions.

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1 A new standalone Section 4.6 is added in CoC Amendment 3 to address thermal analysis of off-normal and accident events. The results are evaluated in Chapter 11.
2 Pressures relative to 1 atm absolute pressure (i.e. gauge pressures) are reported throughout this section.

4.6.2 Accident Events

4.6.2.1 Fire Accidents

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low due to the lack of combustible materials at an ISFSI, a conservative fire event has been assumed and analyzed. The only credible concern is a fire from an on-site transport vehicle fuel tank. Under a postulated fuel tank fire, the outer layers of HI-TRAC or HI-STORM overpacks are heated for the duration of fire by the incident thermal radiation and forced convection heat fluxes. The amount of fuel in the on-site transporter is limited to a volume of 50 gallons.

(a) HI-STORM Fire

The fuel tank fire is conservatively assumed to surround the HI-STORM Overpack. Accordingly, all exposed overpack surfaces are heated by radiation and convection heat transfer from the fire. Based on NUREG-1536 and 10 CFR 71 guidelines [4.6.1], the following fire parameters are assumed:

1. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.

2. The average flame temperature must be at least 1475°F (800°C). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces, which is very conservative considering the size of the HI-STORM cask. It is therefore conservative to use the 1475°F (800°C) temperature.

3. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration.

4. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal measurements [4.6.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft^2×°F) is applied to exposed overpack surfaces during the short-duration fire.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width [4.6.1], the fuel ring surrounding the overpack covers 147.6 ft^2 and has a depth of 0.54 in. From this depth and a constant fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.62 minutes. The fuel consumption rate of 0.15 in/min is a lowerbound value from a Sandia National Laboratories report [4.6.2]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

To evaluate the impact of fire heating of the overpack, a two-dimensional, axisymmetric model of the overpack cylinder was developed with an initial temperature corresponding to normal storage conditions and design heat load. In this model the outer surface and top surface of the overpack were subjected for the duration of fire (3.62 minutes) to the fire conditions defined in this subsection. In the post-fire phase, the overpack cools to an ambient temperature preceding the fire. The transient study is conducted for a period of 5 hours, which is sufficient to allow temperatures in the overpack to reach their maximum values and begin to recede.

Due to the severity of the fire condition radiative heat flux, heat flux from incident solar radiation is negligible and is not included. Furthermore, the smoke plume from the fire would block most of the solar radiation. It is recognized that the ventilation air in contact with the inner surface of the HI-STORM Overpack with design-basis decay heat and normal ambient temperature conditions varies between 80°F at the bottom and 220°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy a miniscule fraction of area of the cylindrical surface of the massive HI-STORM Overpack. Due to the short duration of the fire event and the relative isolation of the ventilation passages from the outside environment, the ventilation air is expected to experience little intrusion of the fire combustion products. As a result of these considerations, it is conservative to assume that the air in the HI-STORM Overpack ventilation passages is held constant at a substantially elevated temperature (300°F) during the entire duration of the fire event.

The thermal transient response of the storage overpack is determined using the ANSYS finite element program. Time-histories for points in the storage overpack are monitored for the duration of the fire and the subsequent post-fire equilibrium phase.

Heat input to the HI-STORM Overpack while it is subjected to the fire is from a combination of an incident radiation and convective heat fluxes to all external surfaces. This can be expressed by the following equation:

$$q_F = h_{fc}(T_A - T_S) + \sigma\varepsilon[(T_A + C)^4 - (T_S + C)^4]$$

where:

- q_F =Surface Heat Input Flux (Btu/ft^2-hr)
- h_{fc} = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft^2-hr-°F)
- σ = Stefan-Boltzmann Constant
- T_A = Fire Temperature (1475°F)
- C= Conversion Constant (460 (°F to °R))
- T_S = Surface Temperature (°F)
- ε = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [4.6.2].

After the fire event, the ambient temperature is restored and the storage overpack cools down (post-fire temperature relaxation). Heat loss from the outer surfaces of the storage overpack is determined by the following equation:

$$q_S = h_S (T_S - T_A) + \sigma\varepsilon [(T_S + C)^4 - (T_A + C)^4]$$

where:

q_S =Surface Heat Loss Flux (W/m^2 (Btu/ft^2-hr))
h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_S = Surface Temperature (°F))
T_A = Ambient Temperature (°F)
σ = Stefan-Boltzmann Constant
ε = Surface Emissivity
C= Conversion Constant (460 (°F to °R))

In the post-fire temperature relaxation phase, h_s is obtained using literature correlations for natural convection heat transfer from heated surfaces [4.2.9].

During the fire the overpack external shell temperatures are substantially elevated (~550°F) and an outer layer of concrete approximately 1 inch thick reaches temperatures in excess of short term temperature limit. This condition is addressed specifically in NUREG-1536 (4.0,V,5.b), which states that:

> "The NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire."

These results demonstrate that the fire accident event analyzed in a most conservative manner is determined to have a minor affect on the HI-STORM Overpack. Localized regions of concrete are exposed to temperatures in excess of accident temperature limit. The bulk of concrete remains below the short term temperature limit. The temperatures of steel structures are within the allowable temperature limits.

Having evaluated the effects of the fire on the overpack, we now evaluate the effects on the MPC and contained fuel assemblies. Guidance for the evaluation of the MPC and its internals during a fire event is provided by NUREG-1536 (4.0,V,5.b), which states:

> "For a fire of very short duration (i.e., less than 10 percent of the thermal time constant of the cask body), the NRC finds it acceptable to calculate the fuel temperature increase by assuming that the cask inner wall is adiabatic. The fuel temperature increase should then be determined by dividing the decay energy released during the fire by the thermal capacity of the basket-fuel assembly combination."

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

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

where:

c_p = Overpack Specific Heat Capacity (Btu/lb-°F)
ρ = Overpack Density (lb/ft³)
L_c = Overpack Characteristic Length (ft)
k = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (142 lb/ft^3) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM Overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 128\,hrs$$

One-tenth of this time constant is approximately 12.8 hours (768 minutes), substantially longer than the fire duration of 3.62 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The fuel temperature rise is computed next.

Table 4.5.2 lists lower-bound thermal inertia values for the MPC and the contained fuel assemblies. Applying a conservative upperbound decay heat load (38 kW ($1.3\text{x}10^5$ Btu/hr)) and adiabatic heating for the 3.62 minutes fire, the fuel temperature rise computes as:

$$\Delta T_{fuel} = \frac{\text{Decay heat} \times \text{Time duration}}{\text{(MPC + Fuel) heat capacities}} = \frac{1.3 \times 10^5 \text{ Btu/hr} \times (3.62/60)hr}{(2240 + 4680) \text{ Btu/}^\circ F} = 1.1^\circ F$$

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

(b) HI-TRAC Fire

In this subsection the fuel cladding and MPC pressure boundary integrity under an exposure to a short duration fire event is demonstrated. The HI-TRAC is initially (before fire) assumed to have a design basis decay heat and has reached steady-state (maximum) temperatures. The analysis assumes a fire from a 50 gallon transporter fuel tank spill. The fuel spill is assumed to surround the HI-TRAC in a 1 m wide ring. The fire parameters are same as that assumed for the HI-STORM fire. Based on the fuel spill defined above the HI-TRAC fire duration is computed as 4.8 minutes.

From the HI-TRAC fire analysis, a bounding MPC temperature rise rate is determined. The total temperature rise (ΔT) is obtained by the product of the rate of temperature rise and fire duration reported above. In this manner the final MPC (fuel and MPC contents) temperature is computed. The MPC pressures are also computed using the MPC temperature rise (ΔT) and Ideal Gas Law. The temperatures and pressures are reported in Table 4.6.7. The results are confirmed to be below the accident temperature and pressure limits (Tables 2.2.3 and 2.2.1).

4.6.2.2 Jacket Water Loss

In this subsection, the fuel cladding and MPC boundary integrity is evaluated for a postulated loss of water from the HI-TRAC water jacket. The HI-TRAC is equipped with an array of water compartments filled with water. For a bounding analysis, all water compartments are assumed to lose their water and be replaced with air. As an additional measure of conservatism, the air in the water jacket is assumed to be motionless (i.e. natural convection neglected) and radiation heat transfer in the water jacket spaces ignored. The HI-TRAC is assumed to have the maximum thermal payload (design heat load) and assumed to have reached steady state (maximum) temperatures. Under these assumed set of adverse conditions, the maximum temperatures are computed and reported in Table 4.6.3. The results of jacket water loss evaluation confirm that the cladding, MPC and HI-TRAC component temperatures are below the limits prescribed in Chapter 2 (Table 2.2.3). The co-incident MPC pressure is also computed and compared with the MPC accident design pressure (Table 2.2.1). The result (Table 4.6.2) is confirmed to be below the limit.

4.6.2.3 Extreme Environmental Temperatures

To evaluate the effect of extreme weather conditions, an extreme ambient temperature (Table 2.2.2) is postulated to persist for a 3-day period. For a conservatively bounding evaluation the extreme temperature is assumed to last for a sufficient duration to allow the HI-STORM 100 System to reach steady state conditions. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative. Starting from a baseline condition evaluated in Section 4.4 (normal ambient temperature and limiting fuel storage configuration) the temperatures of the HI-STORM 100 System are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures (45°F). The HI-STORM extreme ambient temperatures computed in this manner are reported in Table 4.6.4. The co-incident MPC pressure is also computed (Table 4.6.2) and compared with the accident design pressure (Table 2.2.1). The result is confirmed to be below the accident limit.

4.6.2.4 100% Blockage of Air Inlets

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

As a result of the considerable inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for extended durations. This accident condition is, however, a short duration event that is identified and corrected through scheduled periodic surveillance. Nevertheless, this event is conservatively analyzed assuming a substantial duration of blockage. The event is analyzed using the FLUENT CFD code. The CFD model is the same as that constructed for normal storage conditions (see Section 4.4) except for the bottom inlet ducts, which are assumed to be impervious to air. Using this model, a transient thermal solution of the HI-STORM 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.

4.6.2.5 Burial Under Debris

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

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 System will undergo a transient heat up under adiabatic conditions. The minimum available time ($\Delta\tau$) for the fuel cladding to reach the accident limit depends on the following: (i) thermal inertia of the cask, (ii) the cask initial conditions, (iii) the spent nuclear fuel decay heat generation and (iv) the margin between the initial cladding temperature and the accident temperature limit. To obtain a lowerbound on $\Delta\tau$, the HI-STORM 100 Overpack thermal inertia (item i) is understated, the cask initial temperature (item ii) is maximized, decay heat overstated (item iii) and the cladding temperature margin (item iv) is understated. A set of conservatively postulated input parameters for items (i) through (iv) are summarized in Table 4.6.6. Using these parameters $\Delta\tau$ is computed as follows:

$$\Delta\tau = \frac{m \times c_p \times \Delta T}{Q}$$

where:

$\Delta\tau$ = Allowable burial time (hr)
m = Mass of HI-STORM System (lb)
c_p = Specific heat capacity (Btu/lb-°F)
ΔT = Permissible temperature rise (°F)
Q = Decay heat load (Btu/hr)

Substituting the parameters in Table 4.6.6, a substantial burial time (34.6 hrs) is obtained. 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.

Table 4.6.1
OFF-NORMAL CONDITION MAXIMUM
HI-STORM TEMPERATURES[1]

Location[2]	**Off-Normal Ambient Temperature[3] (°F)**	**Partial Inlet Ducts Blockage (°F)**
Fuel Cladding	748	751
MPC Basket	731	734
MPC Shell	500	514
Overpack Inner Shell	376	386
Lid Concrete Bottom Plate	409	408
Lid Concrete Section Temperature	299	291

Table 4.6.2
OFF-NORMAL AND ACCIDENT CONDITION MAXIMUM MPC PRESSURES

Condition	**Pressure (psig)**
Off-Normal Conditions	
Off-Normal Ambient	101.5
Partial Blockage of Inlet Ducts	101.3
Accident Conditions	
Extreme Ambient Temperature	104.5
100% Blockage of Air Inlets	119.3
Burial Under Debris	134.4
HI-TRAC Jacket Water Loss	116.8

1 The temperatures reported in this table are below the off-normal temperature limits specified in Chapter 2, Table 2.2.3.
2 Temperatures of limiting components reported.
3 Obtained by adding the off-normal-to-normal ambient temperature difference of 20°F (11.1°C) to normal condition HI-STORM temperatures reported in Section 4.4.

Table 4.6.3
HI-TRAC JACKET WATER LOSS ACCIDENT MAXIMUM TEMPERATURES

Component	Temperature (°F)
Fuel Cladding	898
MPC Basket	885
MPC Shell	594
HI-TRAC Inner Shell	499
HI-TRAC Enclosure Shell	280

Table 4.6.4

EXTREME ENVIRONMENTAL CONDITION MAXIMUM HI-STORM TEMPERATURES

Component	Temperature[1] (°F)
Fuel Cladding	773
MPC Basket	756
MPC Shell	525
Overpack Inner Shell	401
Lid Concrete Bottom Plate	434
Lid Concrete Section Temperature	324

1 Obtained by adding the extreme ambient to normal temperature difference (45°F) to normal condition temperatures reported in Section 4.4.

Table 4.6.5

SUMMARY OF HI-STORM 100 32-HOURS BLOCKED INLET DUCTS THERMAL ANALYSIS

Component	Initial Temperatures (°F)	Peak Temperatures (°F)
Fuel Cladding	728	916
MPC Basket	711	904
MPC Shell	490	630
Overpack Inner Shell	356	533
Lid Concrete Bottom Plate	389	416
Lid Concrete Section Temperature	279	311

Table 4.6.6

SUMMARY OF INPUTS FOR BURIAL UNDER DEBRIS ANALYSIS

Thermal Inertia Inputs: M (Lowerbound HI-STORM 100 Weight) Cp (Carbon steel heat capacity)[1]	 150000 lb 0.1 Btu/lb-°F
Cask initial temperature	728°F
Q (Decay heat)	1.3×10^5 Btu/hr
ΔT (clad temperature margin)[2]	300°F

1 Carbon steel has the lowest heat capacity among the principal materials employed in MPC and overpack construction (carbon steel, stainless steel and concrete).

2 The clad temperature margin is conservatively understated in this table.

Table 4.6.7

HI-TRAC FIRE ACCIDENT MAXIMUM TEMPERATURES AND PRESSURES

Component	**Initial Temperature (°F)**	**Bounding Temperature Rise (°F)**	**Peak Temperature (°F)**
Fuel Cladding	777	27	804
MPC Shell	507	27	534
Pressures (psig)			
Component	Initial	Pressure Rise	Peak Pressure
MPC	105	3.2	108.2

4.7 REGULATORY COMPLIANCE

4.7.1 Normal Conditions of Storage

NUREG-1536 [4.4.1] and ISG-11 [4.1.4] define several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 4.1 through 4.4. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by ISG-11 [4.1.4], the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System. Maximum clad temperatures for long-term storage conditions are reported in Section 4.4.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Sections 4.4, 4.5 and 4.6 for normal, short term operations, and off-normal & accident conditions. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for long-term storage conditions are reported in Section 4.4. Design temperature limits are summarized in Table 2.2.3. HI-STORM System components defined as important to safety are listed in Table 2.2.6.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 [4.1.4] limit of 400°C (752°F).

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal conditions. All thermal results reported in Section 4.4 are within the design criteria allowable ranges for all normal conditions of storage.

4.7.2 Short Term Operations

Evaluation of short term operations is presented in Section 4.5. This section establishes complete compliance with the provisions of ISG-11 [4.1.4]. In particular, the ISG-11 requirement to ensure that maximum cladding temperatures under all fuel loading and short term operations be below 400°C (752°F) for high burnup fuel and below 570°C (1058°F) for moderate burnup fuel is demonstrated as stated below.

Specifically as required by ISG-11, the fuel cladding temperature is maintained below the applicable limits for HBF and MBF (Table 4.3.1) during short term operations.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal and off-normal conditions, assuming rupture of 1 percent and 10 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods.

As required by NUREG-1536 (4.0,IV, 4), all cask and fuel materials are maintained within their minimum and maximum temperature for all short term operations in order to enable components to perform their intended safety functions.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for all short term operations.

4.8 REFERENCES

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

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

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

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

[4.1.5] "Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data," Holtec Report HI-992252, Revision 1, Holtec International, Marlton, NJ, 08053.

[4.1.6] "Identifying the Appropriate Convection Correlation in FLUENT for Ventilation Air Flow in the HI-STORM System", Holtec Report HI-2043258, Holtec International, Marlton, NJ, 08053.

[4.1.7] "Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask", EPRI TR-100305, (May 1992).

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

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

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

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

[4.2.5] Kern, D.Q., "Process Heat Transfer," McGraw Hill Kogakusha, (1950).

[4.2.6] "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, (August 1981).

[4.2.7] "Spent Nuclear Fuel Effective Thermal Conductivity Report," US DOE Report BBA000000-01717-5705-00010 REV 0, (July 11, 1996).

[4.2.8] ASME Boiler and Pressure Vessel Code, Section II, Part D, (1995).

[4.2.9] Jakob, M. and Hawkins, G.A., "Elements of Heat Transfer," John Wiley & Sons, New York, (1957).

[4.2.10] ASME Steam Tables, 3rd Edition (1977).

[4.2.11] "Nuclear Systems Materials Handbook, Vol. 1, Design Data", ORNL TID 26666.

[4.2.12] "Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year-Old PWR Spent Fuel", ORNL/CSD/TM-149 TTC-0316, (1983).

[4.2.13] "Holtite A: Development History and Thermal Performance Data", Holtec Report HI-2002396, Rev. 3., Holtec International, Marlton, NJ, 08053.

[4.2.14] "Qualification of METAMIC for Spent-Fuel Storage Application", EPRI Report 1003137, (October 2001), EPRI, Palo Alto, CA.

[4.2.15] "Sourcebook for METAMIC Performance Assessment", Holtec Report HI-2043215, Holtec International, Marlton, NJ, 08053.

[4.2.16] USNRC Docket no 72-1027, TN-68 FSAR & Docket no 72-1021 TN-32 FSAR.

[4.2.17] Hagrman, Reymann and Mason, "MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, Tree 1280, Rev. 2, EG&G Idaho, August 1981.

[4.2.18] "Effective Thermal Conductivity and Edge Conductance Model for a Spent-Fuel Assembly," R. D. Manteufel & N. E. Todreas, Nuclear Technology, 105, 421- 440, (March 1994).

[4.3.1] Lanning and Beyer, "Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rode During Short Term Operations for Dry Cask Storage," PNNL White Paper, (January 2004).

[4.4.1] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," USNRC, (January 1997).

[4.4.2] "Pressure Loss Charactersistics for In-Cell Flow of Helium in PWR and BWR Storage Cells", Holtec Report HI-2043285, Holtec International, Marlton, NJ, 08053.

[4.5.1] J.P. Holman, "Heat Transfer," McGraw-Hill Book Company, Sixth Edition, 1986.

[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).

SUPPLEMENT 4.I[1]

THERMAL EVALUATION OF THE HI-STORM 100U SYSTEM

4.I.0 OVERVIEW

The HI-STORM 100U overpack is an underground vertical ventilated module (VVM) with openings for air ingress and egress and internal passages for ventilation cooling of loaded MPCs. The HI-STORM 100U construction is described in Supplement 1.I and illustrated in Figures 1.I.1 through 1.I.3. The HI-STORM 100U inlet and outlets are [redacted]. The overall ventilation airflow path from inlet to outlet is illustrated in Figure 1.I.4. This supplement provides a thermal evaluation of the HI-STORM 100U for normal, off-normal and accident conditions. The evaluations described herein parallel those of the aboveground HI-STORM cask contained in the main body of Chapter 4 of this FSAR. To ensure readability, the section in the main body of the chapter to which each section in this supplement corresponds is clearly identified. All tables in this supplement are labeled sequentially.

4.I.1 INTRODUCTION

The information presented in this supplement is intended to serve as a complement to the information provided in the main body of Chapter 4. Thus, information in Chapter 4 that remains applicable to the HI-STORM 100U is not repeated. Specifically, the following information in the main body of Chapter 4 is not repeated herein:

1. The thermal properties of materials in Section 4.2 that are applicable to the HI-STORM 100U System.

2. The specifications for components in Section 4.3 that are applicable to the HI-STORM 100U System.

3. The descriptions of the thermal modeling of the MPC and its internals, including fuel assemblies, in Section 4.4 are applicable in their entirety to the HI-STORM 100U.

4. The descriptions of the short-term loading operations, carried out using the HI-TRAC transfer cask, in Section 4.5 remain applicable in their entirety to the HI-STORM 100U.

As demonstrated by appropriate supporting analyses, the heat rejection capability of the HI-STORM 100U System is essentially equivalent to its aboveground counterparts (strictly speaking, slightly better, because of the larger intake and outlet passages located in the VVM lid). Further, its underground configuration renders its resistance to accident events such as fire greater than that of aboveground casks.

[1] For ease of supplement review the sections are numbered in parallel with the main Chapter 4.

4.I.2 THERMAL PROPERTIES OF MATERIALS[1]

The material properties compiled in Section 4.2 of the FSAR provide the required materials information, [redacted]

4.I.3 SPECIFICATIONS FOR COMPONENTS[2]

All applicable material temperature limits in Section 4.3 of the FSAR continue to apply to the HI-STORM 100U. [redacted]

4.I.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE[3]

4.I.4.1 HI-STORM 100U Thermal Model

The thermal model of the HI-STORM 100U overpack is constructed on the FLUENT CFD program [4.1.2], and has the following key attributes:

1. The airflow through the cooling passages of the VVM is modeled as turbulent, using the k-ω model with transitional option as recommended in a Holtec-proprietary benchmarking report [4.1.6]. This is the same modeling approach as used in the aboveground cask analyses.

2. The MPC is modeled as two axisymmetric zones as shown in Figure 4.4.2 for the aboveground cask analyses in Section 4.4 with bottom and top plenums for modeling internal convection. The use of the axisymmetric model for the MPC has been demonstrated to be conservative via comparison with three-dimensional models in Section 4.4.

[1] This section supplements Section 4.2.
[2] This section supplements Section 4.3.
[3] This section supplements Section 4.4.

3. The helium flow within the MPC is modeled as laminar. This is the same modeling approach used in the aboveground cask analyses.

4. The hydraulic resistance of the fuel assemblies stored within the MPC is obtained using 3-D CFD models of design-basis assemblies specified in Chapter 2. The hydraulic resistance calculations are provided in a Holtec proprietary calculation package [4.4.2]. This is the same as used in the HI-STORM aboveground modeling.

5. The vertical surfaces between adjacent modules are assumed insulated.

6. The underside of the VVM foundation pad (see Figure 1.I.1) is assumed supported on a subgrade at 77°F. This is the same boundary condition applied to the bottom of the ISFSI pad for the aboveground cask modeling in Section 4.4.

7. Because the wind renders the problem non-axisymmetric, a 3-D half cylinder extension of the axisymmetric model of the VVM is required. [redacted]

The VVM thermal model is constructed using the same modeling platform used for aboveground analysis (FLUENT version 6.1).

4.I.4.2 Thermal Analysis

The HI-STORM 100U System design has been developed with the objective of ensuring that its thermal performance is sufficient to meet allowable temperature and pressure limits (defined in Chapter 2 and Supplement 2.I). Further, it is desired that the underground overpack meet or exceed the thermal performance of the aboveground HI-STORM systems. In this supplement the HI-STORM 100U System is evaluated to demonstrate compliance with specified temperature and pressure limits and to confirm its heat rejection capacity meets or exceeds that of the aboveground cask systems. All HI-STORM overpack designs utilize the same MPCs.

To demonstrate that the HI-STORM 100U thermal performance meets the regulations, a steady-state thermal analysis of a HI-STORM 100U containing an MPC-32 is performed for normal conditions of long-term storage for an array of regionalized and uniform storage configurations defined in Chapter 2, Section 2.1. Other conditions (such as ambient temperature, helium operating pressure and insolation) are same as for the aboveground overpack. As all MPCs have identical outer dimensions, the geometry of the cooling airflow passages in the HI-STORM 100U are identical for all MPC designs and obtaining satisfactory results for any MPC design ensures satisfactory results for all designs.

In Table 4.I.8 peak cladding temperature results in the aboveground and underground HI-STORM overpacks is provided for the range of regionalized loading parameter X specified in

Chapter 2 ($0.5 \leq X \leq 3$). From this table it is evident that the underground overpack temperatures closely follows the temperatures in the aboveground overpack from below. Table 4.I.2 presents the HI-STORM 100U temperatures for a limiting storage condition wherein the highest clad temperatures are reached ($X = 0.5$). As this condition is co-incident with the maximum permissible MPC heat load, the other system temperatures (such as fuel basket, MPC shell, overpack and air temperatures) also reach their highest values. Accordingly, this condition is utilized for all off-normal and accident conditions evaluated in this supplement.

In accordance with Supplement 2.I, a normal wind (i.e., non-quiescent) condition is also evaluated. This scenario is identical to that described above, except for a horizontal wind of up to 10 miles per hour. It should be recognized that these evaluations assume a wind of sufficient duration for the HI-STORM 100U System to reach a steady thermal state, which is an extremely conservative assumption given the large thermal inertia of the system. Results of wind evaluation are presented in Table 4.I.9 and cladding temperatures confirmed to be below the allowable long-term normal temperature limits.

The results of these normal condition analyses demonstrate that the HI-STORM 100U System will yield fuel cladding, MPC and VVM temperatures well below the allowable limits and that the MPC internal pressure is also below its allowable limit. These results confirm the safety of the HI-STORM 100U System.

4.I.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

The short term evaluations presented in Section 4.5 are applicable in their entirety for the underground overpack.

4.I.6 THERMAL EVALUATION OF OFF-NORMAL AND ACCIDENT CONDITIONS[1]

4.I.6.1 Off-Normal Conditions

(a) Elevated Ambient Air Temperature

The elevated ambient air temperature off-normal condition is defined in Table 2.I.1 as an ambient temperature of 100°F. This is 20°F higher than the normal condition ambient temperature of 80°F, also defined in Table 2.I.1 and used in the analyses described in Section 4.I.4.2 above. This condition is conservatively evaluated by adding 20°F to the calculated normal condition fuel cladding and component temperatures. Results for this off-normal condition are presented in Table 4.I.3. The results are confirmed to be less than short-term temperature limits for fuel cladding, concrete, and ASME Code materials.

(b) Partial Blockage of Air Inlets

[1] This section supplements Section 4.6.

The partial air inlets blockage event is defined in Table 2.I.1 as the blockage of 50% of the air inlet flow area. This event is evaluated by blocking one half of the inlet opening and computing the asymptotic (i.e. bounding steady-state) temperature field. Results for this off-normal condition are presented in Table 4.I.4. The results are confirmed to be less than short-term temperature limits for fuel cladding, concrete, and ASME Code materials.

(c) Elevated Wind Speed

The elevated wind speed event is defined in Table 2.I.1 as a horizontal wind speed of up to 30 miles per hour. This event is evaluated using the thermal model described in Section 4.I.4.2 with a horizontal wind blowing into the inlet openings from one direction. Because the 100U ventilation openings are axisymmetric the effect of wind on the inlets is the same regardless of direction. It should be recognized that this evaluation assumes unidirectional wind of sufficient duration for the HI-STORM 100U System to reach a steady thermal state, which is a conservative assumption given the large thermal inertia of system. Results for this off-normal condition are presented in Table 4.I.9 and cladding temperatures confirmed to remain well below the short-term allowable limit.

4.I.6.2 Accident Conditions

(a) Fire

The fire accident is defined in Table 2.I.1 as a 1475°F fire lasting 217 seconds. This is the same intensity and duration as the fire accident evaluated in Section 4.6 of this FSAR for the aboveground overpacks. The existing fire evaluation therein will bound the HI-STORM 100U fire event, for the following reasons:

1. Because the fire evaluated in Section 4.6 is an engulfing fire, the cask area exposed to the fire heat flux is maximized. The underground surfaces of the HI-STORM 100U VVM are not directly exposed to the fire heat flux, which significantly reduces the fire heat input to the VVM as compared to an aboveground overpack. The total heat input to the VVM during the fire event is therefore much lower than is evaluated in Section 4.6.

2. The openings of the inlet ducts and outlet ducts are both located near the top of the VVM. Because heated gases rise, a downward intrusion of combustion gases into the module cavity is not credible. The internal surfaces of the VVM cannot, therefore, be subjected to any significant temperature elevation due to fire.

Based on these considerations, the fire evaluation for the aboveground overpack bounds the HI-STORM 100U fire accident.

(b) Flood

The flood accident is defined in Table 2.I.1 as a deep submergence. The worst flood from a thermal perspective is a "smart flood" that just prevents all airflow with *no* MPC cooling by water. Although the HI-STORM 100U includes design features to prevent "smart flood" occurrence such a hypothetical condition is bounded by the 100% inlet ducts blocked accident evaluated in 4.I.4.2(d). As shown in the HI-STORM 100U licensing drawings, the bottom of the MPC is situated [redacted] the MPC baseplate heat dissipation ensures adequate cooling of the MPC and its stored fuel. This effect is significant, because the thermosiphon convective flow within the MPC is an efficient means of heat rejection to the thermal sink (wetted baseplate). To illustrate this a *steady state* flood condition is postulated wherein the bulk of the airflow plenum (upto MPC baseplate) is assumed to be flooded. The results of this condition are presented in Table 4.I.5. All cladding, concrete and metal temperatures are below the short-term limits.

(c) Burial Under Debris

The burial under debris accident is defined in Table 2.I.1 as an adiabatic heat-up at the maximum decay heat load. The existing burial under debris evaluation in Section 4.6 will bound the HI-STORM 100U burial under debris event because the HI-STORM 100U System thermal inertia is greater than that of the aboveground systems. This results from the higher aggregate mass of the VVM top concrete and subgrade backfill as compared to the aboveground overpacks. As such, it is evident that the existing burial under debris evaluation for the aboveground overpack will bound the HI-STORM 100U burial under debris accident as well.

(d) 100% Blockage of Air Ducts

The 100% air ducts blockage accident is defined in Table 2.I.1 as the blockage of 100% of the air inlet duct flow area. This event is evaluated by blocking the entire inlet opening for a considerable duration (26 hours) and performing a transient calculation of overpack, MPC and cladding temperatures. The only difference between this evaluation and the evaluation described in Section 4.I.3.2 is the blockage of the inlet vents and the inclusion of transient effects. Numeric results for this accident are presented in Table 4.I.6. The results demonstrate that all fuel cladding and component temperatures remain below their respective short-term limits.

It should be noted that the increase in temperature would increase the MPC internal pressure. The calculation performed for this accident recognizes an increase in thermosiphon cooling within the MPC that would accompany from pressure increase in a conservative manner.

(e) Extreme Environmental Temperature

The extreme environmental temperature accident condition is defined in Table 2.I.1 as an ambient temperature of 125°F. This is 45°F higher than the normal condition ambient temperature of 80°F, also defined in Table 2.I.1 and used in the analyses described in Section 4.I.3.2 above. This condition is conservatively evaluated by adding 45°F to the calculated normal condition fuel cladding and component temperatures. Results for this off-normal condition are presented in Table 4.I.7. The results are confirmed to be less than accident temperature limits for fuel cladding, concrete, and ASME Code materials.

It should be noted that an increase in temperature is followed by a concomitant increase in MPC helium pressure. The bounding calculation performed for this accident does not take any credit for the increase in thermosiphon cooling within the MPC that would accompany the pressure increase. As an increase in thermosiphon cooling would limit the temperature rise resulting from an elevated ambient temperature, the calculated temperatures and pressures are bounding.

4.1.7 REGULATORY COMPLIANCE

As required by ISG-11, the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM System.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask materials and fuel cladding are maintained within their temperature limits for normal, off-normal and accident conditions in order to enable components to perform their intended safety functions. Material temperature limits are summarized in Tables 2.2.3 and 2.I.7. HI-STORM 100U System components defined as important to safety are listed in Tables 2.2.6 and 2.I.7.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 limit of 400°C (752°F).

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this supplement, including conduction, natural convection, and thermal radiation, are passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in Chapters 2 and 3 for normal conditions. All thermal results are within the allowable limits for all normal conditions of storage.

Table 4.I.1

[redacted]

Thermal Property	Specified Value
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

Table 4.I.2

HI-STORM 100U System Normal Storage Maximum Temperatures[2] and Pressures

Component	Temperature (°F)
Fuel Cladding	711
Fuel Basket	663
Fuel Basket Periphery	561
MPC Shell	479
MPC Lid	490
VVM Container Shell	127
VVM Lid Bottom Plate	328
Lid Concrete	258
Average Air Outlet Temperature	185
[redacted]	[redacted]
Pressure (psig)	
MPC	98

[redacted]

[2] The axially averaged temperature of the [redacted] does not exceed 300°F.

Table 4.I.3

Results under Off-Normal Ambient Temperature

Component	Temperature (°F)
Fuel Cladding	731
Fuel Basket	683
Fuel Basket Periphery	581
MPC Shell	499
MPC Lid	510
VVM Container Shell	147
VVM Lid Bottom Plate	348
VVM Lid Concrete	278
[illegible]	[illegible]
Pressure (psig)	
MPC	103

Table 4.I.4

Results for Partial Blockage of Air Inlets

Component	Temperature (°F)
Fuel Cladding	758
Fuel Basket	709
Fuel Basket Periphery	591
MPC Shell	511
MPC Lid	528
VVM Container Shell	147
VVM Lid Bottom Plate	374
VVM Lid Concrete	296
[illegible]	[illegible]
Pressure (psig)	
MPC	103

Table 4.I.5

Results for Flood Accident

Component	Temperature (°F)
Fuel Cladding	719
Fuel Basket	670
Fuel Basket Periphery	565
MPC Shell	483
MPC Lid	495
VVM Container Shell	131
VVM Lid Bottom Plate	336
VVM Lid Concrete	265
[illegible]	[illegible]
Pressure (psig)	
MPC	99

Table 4.I.6

Results of a 26-Hour 100% Air Inlets Blockage Accident

Component	Temperature (°F)
Fuel Cladding	1048
Fuel Basket	1000
Fuel Basket Periphery	856
MPC Shell	807
MPC Lid	624
VVM Container Shell	296
VVM Lid Bottom Plate	456
VVM Lid Concrete	330
[illegible]	[illegible]
Pressure (psig)	
MPC	138

Table 4.I.7

Results for Extreme Environmental Temperature Accident

Component	Temperature (°F)
Fuel Cladding	756
Fuel Basket	708
Fuel Basket Periphery	606
MPC Shell	524
MPC Lid	535
VVM Container Shell	172
VVM Lid Bottom Plate	373
VVM Lid Concrete	303
[illegible]	[illegible]
Pressure (psig)	
MPC	108

Table 4.I.8

Comparison of Peak Cladding Temperatures in Aboveground and Underground Overpacks

X (Regionalized Loading Parameter)	Aboveground case Peak Clad Temperature (°F)	Underground case Peak Clad Temperature (°F)
0.5	720	711
1.0	715	704
2.0	716	702
3.0	713	701

Table 4.I.9

Effect of Wind on Peak Cladding Temperatures

Wind Speed	Peak Cladding Temperature (°F)
Normal Wind	
2.5	721
5	717
10	647
Off-Normal Wind	
20	584
30	518

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 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) prior to being loaded into the MPC. ~~DFCs containing BWR fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing BWR damaged fuel assemblies may be stored in either the MPC-68, the MPC-68F, or the MPC-68FF. DFCs containing PWR fuel debris must be stored in the~~

† 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 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.~~

~~MPC-24EF or MPC-32F while DFCs containing PWR damaged fuel assemblies may be stored in either the MPC-24E, MPC-24EF, MPC-32, or MPC-32F.~~

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

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs) or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices.

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 System offers two different loading patterns in the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, and the MPC-68FF. These patters are uniform and regionalized loading as described in Section 2.0.1 and 2.1.6. Since the different loading patterns have different allowable burnup and cooling times combinations, both loading patterns are discussed in this chapter.

The sections that follow will demonstrate that the design of the HI-STORM 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR 72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- The HI-STORM 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STORM 100 System is acceptable during normal, off-normal, and accident conditions.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains a discussion on the release of radioactive materials from the HI-STORM 100 System. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM 100 System.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

5.1 DISCUSSION AND RESULTS

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

- Gamma radiation originating from the following sources

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

- Neutron radiation originating from the following sources

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

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

The shielding analyses were performed with MCNP-4A [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.3 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6 and mixed oxide (MOX) fuel assemblies are the GE 6x6. The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes. Section 2.1.9 specifies the acceptable intact zircaloy clad fuel characteristics and the acceptable damaged fuel characteristics.

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

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

Section 2.1.9 specifies that the maximum assembly average burnup for PWR and BWR fuel is 68,200 and 65,000 MWD/MTU, respectively. The analysis in this chapter conservatively considers burnups up to 75,000 and 70,000 MWD/MTU for PWR and BWR fuel, respectively. ~~The burnup and cooling times used in this chapter were conservatively chosen to bound burnup and cooling times based on assembly decay heat values of 1.583, 1.1875, and 0.522 kW for the MPC-24, MPC-32, and MPC-68, respectively. These decay heat values bound those reported in Section 2.1.9.~~

~~The dose rates surrounding the HI-STORM overpack are very low, and thus, the shielding analysis of the HI-STORM overpack conservatively considered the burnup and cooling time combinations listed below, which bound the acceptable burnup levels and cooling times from Section 2.1.9. This large conservatism is included in the analysis of the HI-STORM overpack to unequivocally demonstrate that the HI-STORM overpack meets the Part 72 dose requirements.~~*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**
~~47,500~~*60,000* MWD/MTU 3 year cooling	~~35,000~~*45,000* MWD/MTU 3 year cooling	~~40~~*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*

Zircaloy Clad Fuel		
MPC-24	**MPC-32**	**MPC-68**
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
Stainless Steel Clad Fuel		
MPC-24	**MPC-32**	**MPC-68**
40,000 MWD/MTU 8 year cooling	40,000 MWD/MTU 9 year cooling	22,500 MWD/MTU 10 year cooling

Results are presented in this chapter for the single burnup and cooling time combination for zircaloy clad fuel from the above table which produces the highest dose rate at 1 meter from the midplane of the HI-STORM overpack and HI-TRAC transfer casks. The burnup and cooling time combination may be different for normal and accident conditions and for the different overpacks. ~~The burnup and cooling time combinations analyzed for zircaloy clad fuel produce dose rates at the midplane of the HI-STORM overpack which bound all uniform and regionalized loading burnup and cooling time combinations listed in Section 2.1.9. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.~~

~~The dose rates surrounding the HI-TRAC transfer cask are significantly higher than the dose rates surrounding the HI-STORM overpack, and although no specific regulatory limits are defined, dose rates are based on the ALARA principle. Therefore, the cited dose rates were based on burnups and cooling times closer to the combinations in Section 2.1.9. Two different burnup and cooling times, listed below, were analyzed for the MPC-24, MPC-32, and the MPC-68 in the 100-ton HI-TRAC. The burnup and cooling time combinations were chosen for the minimum cooling time and a bounding burnup corresponding to the 14x14A in the MPC-24 and MPC-32 and the 9x9G fuel assembly in the MPC-68. The burnups corresponding to 3-year cooling times produce dose rates at 1 meter from the radial surface of the overpack, for the locations reported in this chapter, which bound the dose rates from all other uniform loading burnup and cooling time combinations in Section 2.1.9.~~

~~100-ton HI-TRAC~~		
~~**MPC-24**~~	~~**MPC-32**~~	~~**MPC-68**~~
~~46,000 MWD/MTU 3 year cooling~~	~~35,000 MWD/MTU 3 year cooling~~	~~39,000 MWD/MTU 3 year cooling~~
~~75,000 MWD/MTU~~	~~75,000 MWD/MTU~~	~~70,000 MWD/MTU~~

~~5-year cooling~~	~~8-year cooling~~	~~6-year cooling~~

As mentioned earlier, there are different versions of the HI-STORM overpack: the HI-STORM 100, the HI-STORM 100S, and the HI-STORM 100S Version B. Section 5.3 describes all three overpacks. However, since the HI-STORM 100S Version B overpack has higher dose rates at the inlet vents and slightly higher offsite dose rates than the other overpacks, results are only presented for the HI-STORM 100S Version B overpack.

The 100-ton HI-TRAC with the MPC-24 has higher dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results ~~for 3-year cooling~~ are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The MPC-32 results, MPC-68 results, and additional MPC-24 results are provided in Section 5.4 for comparison.

The 100-ton HI-TRAC dose rates bound the HI-TRAC 125 and 125D dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the HI-TRAC 125 and 125D. Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. Therefore, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.~~ Dose rates are presented for two different burnup and cooling time combinations for the MPC-24 in the HI-TRAC 125D which bound the allowable contents in Section 2.1.9: 46,000 MWD/MTU with 3-year cooling and 75,000 MWD/MTU with 5-year cooling. The dose rates for the later combination are presented in this section because it produces the highest dose rate at the cask midplane. Dose rates for the other burnup and cooling time combination are presented in Section 5.4.~~

As a general statement, the dose rates for uniform loading presented in this chapter bound the dose rates for regionalized loading~~ at 1 meter distance from the overpack~~. Therefore, dose rates for specific burnup and cooling time combinations in a regionalized loading pattern are not presented in this chapter. Section 5.4.9 provides an additional brief discussion on regionalized loading.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.

2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM 100 System in Section 2.3.5.2 as: ~~135~~ *300* mrem/hour on the radial surface of the overpack, ~~135~~ *175* mrem/hour at the openings of the air vents, and 60 mrem/hour on the top of the overpack.

The HI-STORM overpack dose rates presented in this section are conservatively evaluated for the MPC-32, the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in Section 2.1.9.

Figures ~~5.1.1, 5.1.12, and~~ 5.1.13 identif~~y~~*ies* the locations of the dose points referenced in the dose rate summary tables for the ~~HI-STORM 100, HI-STORM 100S, and~~ HI-STORM 100S Version B overpack~~s, respectively~~. Dose Points # 1 and # 3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point # 4 is the peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. ~~Dose Point # 4a in Figure 5.1.12 is located directly above the exit duct and next to the concrete shield block.~~ The dose values reported at the locations shown on Figures ~~5.1.1, 5.1.12, and~~ 5.1.13 are averaged over a region that is approximately 1 foot in width.

The total dose rates presented in this chapter for the MPC-24 and MPC-32 are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Section 5.2.4.1. This is conservative because it is not expected that the cooling times for both the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values.

~~Tables 5.1.1 and 5.1.3 provide the maximum dose rates adjacent to the HI-STORM 100S overpack during normal conditions for the MPC-32 and MPC-68. Tables 5.1.4 and 5.1.6 provide the maximum dose rates at one meter from the HI-STORM 100S overpack. Tables 5.1.2 and 5.1.5 provide the maximum dose rates adjacent to and one meter from the HI-STORM 100 overpack for the MPC-24.~~

Tables 5.1.11, 5.1.12, and 5.1.13 provide the maximum dose rates adjacent to the HI-STORM 100S Version B overpack during normal conditions for the MPC-32, MPC-24, and MPC-68.

Tables 5.1.14 through 5.1.16 provide the maximum dose rates at one meter from the HI-STORM 100S Version B overpack.

~~Both the HI-STORM 100 and~~*The* HI-STORM 100S Version B overpacks ~~were~~ *was* analyzed for the dose rate at the controlled area boundary. Although the dose rates for the MPC-32 in HI-STORM 100S *Version B* are greater than those for the MPC-24 in HI-STORM 100*S Version B* at the ventilation ducts, as shown in Tables ~~5.1.1, 5.1.2, 5.1.4, and 5.1.5~~*5.1.11 and 5.1.12*, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary for the HI-STORM 100*S Version B* overpack. This is acceptable because the vents are a small fraction of the radial surface area *and the MPC-24 has higher dose rates at the radial midplane than the MPC-32 in the HI-STORM 100S Version B overpack.* ~~As such, the dominant effect on the dose at distance is the radial portion of the overpack between the vents which comprises approximately 91% of the total radial surface area compared to approximately 1.3% for the vents. The MPC-24 was also used for the dose rates at the controlled area boundary from the HI-STORM 100S Version B overpack.~~ The MPC-24 was *also* chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the MPC-32 or the MPC-68 (see Section 2.1.9). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32. ~~The dose rates at the controlled area boundary were calculated for the HI-STORM 100 and HI-STORM 100S Version B overpacks rather than the HI-STORM 100S overpack. The difference in height will have little impact on the dose rates at the controlled area boundary since the surface dose rates are very similar.~~ The controlled area boundary dose rates were also calculated including the BPRA non-fuel hardware source. In the site specific dose analysis, users should perform an analysis which properly bounds the fuel to be stored including BPRAs if present.

Table 5.1.7 provides dose rates adjacent to and one meter from the 100-ton HI-TRAC. Table 5.1.8 provides dose rates adjacent to and one meter from the 125-ton HI-TRACs. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125 and 100 transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point # 5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed burnup and cooling times specified in Section 2.1.9, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen to bound the allowable burnup and cooling times in Section 2.1.9. Results for other burnup and cooling times and for the MPC-68 and MPC-32 are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRACs or the HI-STORM overpack, it is important to understand the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. The figures discussed below were all calculated without the gamma source from BPRAs and were calculated for an earlier design of the HI-TRAC which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below.

Figure 5.1.5 shows the dose rate profile at 1 foot from the side of the 100-ton HI-TRAC transfer cask with the MPC-24 for 35,000 MWD/MTU and 5 year cooling. This figure clearly shows the behavior of the total dose rate and each of the dose components as a function of the cask height. To capture the effect of scattering off the concrete floor, the calculational model simulates the 100-ton HI-TRAC at a height of 6 inches (the typical cask carry height) above the concrete floor. As expected, the total dose rate on the side near the top and bottom is dominated by the Co-60 gamma dose component, while the center dose rate is dominated by the fuel gamma dose component.

The total dose rate and individual dose rate components on the surface of the pool lid on the 100-ton HI-TRAC are provided in Figure 5.1.6, illustrating the significant reduction in dose rate with increasing distance from the center of the pool lid. Specifically, the total dose rate is shown to drop by a factor of more than 20 from the center of the pool lid to the outer edge of the HI-TRAC. Therefore, even though the dose rate in Table 5.1.7 at the center of the pool lid is substantial, the dose rate contribution, from the pool lid, to the personnel exposure is minimal.

The behavior of the dose rate 1-foot from the transfer lid is shown in Figure 5.1.7. Similarly, the total dose rate and the individual dose rate components 1-foot from the top lid, as a function of distance from the axis of the 100-ton HI-TRAC, are shown in Figure 5.1.8. For both lids (transfer and top), the reduction in dose rate with increased distance from the cask axial centerline is substantial.

To reduce the dose rate above the water jacket, a localized temporary shield ring, described in Chapter 8, may be employed on the 125-ton HI-TRACs and on the 100-ton HI-TRAC. This temporary shielding, which is water, essentially extends the water jacket to the top of the HI-TRAC. The effect of the temporary shielding on the side dose rate above the water jacket (in the area around the lifting trunnions and the upper flange) is shown on Figure 5.1.9, which shows the dose profile on the side of the 100-ton HI-TRAC with the temporary shielding installed. For comparison, the total dose rate without temporary shielding installed is also shown on Figure 5.1.9. The results indicate that the temporary shielding reduces the dose rate by approximately a factor of 2 in the area above the water jacket.

To illustrate the reduction in dose rate with distance from the side of the 100-ton HI-TRAC, Figure 5.1.10 shows the total dose rate on the surface and at distances of 1-foot and 1-meter.

Figure 5.1.11 plots the total dose rate at various distances from the bottom of the transfer lid, including distances of 1, 5, 10, and 15 feet. Near the transfer lid, the total dose rate is shown to decrease significantly as a function of distance from the 100-ton HI-TRAC axial centerline. Near the axis of the HI-TRAC, the reduction in dose rate from the 1-foot distance to the 15-foot distance is approximately a factor of 15. The dose rate beyond the radial edge of the HI-TRAC is also shown to be relatively low at all distances from the HI-TRAC transfer lid. Thus, prudent transfer operating procedures will employ the use of distance to reduce personnel exposure. In addition, when the HI-TRAC is in the horizontal position and is being transported on site, a missile shield may be positioned in front of the HI-TRAC transfer lid or pool lid. If present, this shield would also serve as temporary gamma shielding which would greatly reduce the dose rate in the vicinity of the transfer lid or pool lid. For example, if the missile shield was a 2 inch thick steel plate, the gamma dose rate would be reduced by approximately 90%.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single ~~HI-STORM 100 cask and a single~~ HI-STORM 100S Version B cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. These values were conservatively calculated for a burnup of ~~47,500~~*60,000* MWD/MTU and a 3-year cooling time. In addition, the annual dose was calculated for *a* burnup~~s~~ of 45,000 ~~and 52,500~~ MWD/MTU with *a* corresponding cooling time~~s~~ of 9 ~~and 5~~ years ~~respectively~~. BPRAs were included in these dose estimates. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212~~, as stated in Chapter 12, "Operating Controls and Limits"~~. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM 100 cask array configurations provided in Table 5.1.9. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis fuels. Since the source strengths of the GE 6x6 intact and damaged fuel and the GE 6x6 MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the dose rates from the GE 6x6 fuels for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either GE 6x6 intact or damaged or GE 6x6 MOX fuel for normal conditions is required to demonstrate that the MPC-68 with GE 6x6 fuels will meet the normal condition regulatory requirements. Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E, MPC-32 and the MPC-68.

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM 100 System. Section 5.4.6 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 lists the gamma and neutron sources for the design basis stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.4.

The analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.104 limits and ALARA practices.

5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 Rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled are shall be at least 100 meters.

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

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

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

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point # 2) adjacent to the HI-TRAC. Loss ofthe neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point # 2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Scaling this accident dose rate by the dose rate reduction seen in HI-STORM yields a dose rate at the 100 meter controlled area boundary that would be approximately ~~4.28~~*4.22*[†] mrem/hr for the HI-TRAC accident condition. At this dose rate, it would take ~~1168~~*1185* hours (~~~48~~*49* days) for the dose at the controlled area boundary to reach 5 Rem. Assuming a 30 day accident duration, the accumulated dose at the controlled area boundary would be ~~3.08~~*3.04* Rem. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

The consequences of the design basis accident conditions for the MPC-68 and MPC-24E storing damaged fuel ~~and the MPC-68F, MPC-68FF, or MPC-24EF storing damaged fuel~~ and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. Analyses in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition does not significantly affect the dose rates around the cask. Therefore, the damaged fuel post-accident dose rates are bounded by the intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.106 limits.

† 5927.95 mrem/hr (Table 5.1.10) x [~~349.53~~*880.47* mrem/yr (Table 5.4.7) / 8760 hrs / ~~55.26~~*141.21* mrem/hr (Table 5.1.15)]

Table 5.1.1

DELETED

~~DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING~~
~~BURNUP AND COOLING TIME~~
~~35,000 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~† ~~Location~~	~~Fuel Gammas~~†† ~~(mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~1~~	~~15.43~~	~~18.32~~	~~3.24~~	~~36.99~~	~~37.94~~
~~2~~	~~85.14~~†††	~~0.05~~	~~1.10~~	~~86.30~~	~~92.53~~
~~3~~	~~16.04~~	~~18.95~~	~~2.92~~	~~37.92~~	~~46.16~~
4	~~3.24~~	~~1.18~~	~~0.95~~	~~5.37~~	~~6.12~~
4a	~~7.20~~	~~10.46~~	~~13.87~~	~~31.53~~	~~36.41~~

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 4.1 % of this dose rate.

Table 5.1.2

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~~DOSE RATES ADJACENT TO HI-STORM 100 OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING~~
~~BURNUP AND COOLING TIME~~
~~47,500 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~† ~~Location~~	~~Fuel Gammas~~†† ~~(mrem/hr)~~	^{60}~~Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~1~~	~~11.14~~	~~6.61~~	~~3.70~~	~~21.46~~	~~21.84~~
~~2~~	~~88.86~~†††	~~0.04~~	~~2.52~~	~~91.41~~	~~96.85~~
~~3~~	~~7.51~~	~~4.36~~	~~1.84~~	~~13.71~~	~~15.38~~
4	~~1.74~~	~~0.49~~	~~4.82~~	~~7.05~~	~~7.51~~

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 4 % of this dose rate.

Table 5.1.3

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~~DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK FOR NORMAL CONDITIONS~~
~~MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING BURNUP AND COOLING TIME~~
~~40,000 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~[†] ~~Location~~	~~Fuel Gammas~~[††] ~~(mrem/hr)~~	^{60}~~Co Gammas~~ ~~(mrem/hr)~~	~~Neutrons~~ ~~(mrem/hr)~~	~~Totals~~ ~~(mrem/hr)~~
~~1~~	~~15.31~~	~~14.43~~	~~5.79~~	~~35.53~~
~~2~~	~~77.57~~	~~0.01~~	~~1.79~~	~~79.37~~
~~3~~	~~6.63~~	~~21.89~~	~~2.58~~	~~31.10~~
~~4~~	~~1.83~~	~~1.58~~	~~0.99~~	~~4.40~~
~~4a~~	~~1.99~~	~~15.20~~	~~13.46~~	~~30.65~~

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.4

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~~DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING~~
~~BURNUP AND COOLING TIME~~
~~35,000 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~† ~~Location~~	~~Fuel Gammas~~†† ~~(mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~1~~	~~10.64~~	~~6.18~~	~~0.47~~	~~17.29~~	~~18.11~~
~~2~~	~~44.43~~†††	~~0.40~~	~~0.43~~	~~45.26~~	~~48.50~~
~~3~~	~~8.32~~	~~5.33~~	~~0.46~~	~~14.12~~	~~16.82~~
4	~~0.83~~	~~0.37~~	~~0.42~~	~~1.62~~	~~1.84~~

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 4.1 % of this dose rate.

Table 5.1.5

DELETED

~~DOSE RATES AT ONE METER FROM HI-STORM 100 OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING~~
~~BURNUP AND COOLING TIME~~
~~47,500 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~† ~~Location~~	~~Fuel Gammas~~†† ~~(mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~1~~	~~11.15~~	~~3.94~~	~~0.72~~	~~15.82~~	~~16.36~~
~~2~~	~~46.78~~†††	~~0.33~~	~~1.04~~	~~48.16~~	~~50.95~~
~~3~~	~~6.51~~	~~2.84~~	~~0.28~~	~~9.64~~	~~10.87~~
~~4~~	~~0.84~~	~~0.22~~	~~1.47~~	~~2.53~~	~~2.66~~

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 4 % of this dose rate.

Table 5.1.6

DELETED

~~DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING~~
~~BURNUP AND COOLING TIME~~
~~40,000 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point~~† ~~Location~~	~~Fuel Gammas~~†† ~~(mrem/hr)~~	^{60}~~Co Gammas~~ ~~(mrem/hr)~~	~~Neutrons~~ ~~(mrem/hr)~~	~~Totals~~ ~~(mrem/hr)~~
~~1~~	~~10.70~~	~~4.55~~	~~0.78~~	~~16.03~~
~~2~~	~~39.27~~	~~0.32~~	~~0.74~~	~~40.33~~
~~3~~	~~4.38~~	~~6.36~~	~~0.33~~	~~11.07~~
4	~~0.47~~	~~0.50~~	~~0.44~~	~~1.41~~

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
~~46~~*60*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*124.94*	*33.16*	*958.00*	*469.38*	*1585.48*	*1594.02*
2	*3196.67*†	*134.95*	*0.96*	*249.02*	*3581.60*	*3828.84*
3	*36.47*	*6.50*	*528.22*	*392.74*	*963.94*	*1112.46*
3 (temp)	*16.31*	*11.57*	*244.83*	*6.31*	*279.02*	*347.16*
4	*80.38*	*2.57*	*425.12*	*483.50*	*991.57*	*1116.07*
4 (outer)	*23.94*	*1.63*	*105.85*	*326.35*	*457.77*	*489.10*
5 (pool lid)	*623.98*	*47.34*	*4826.78*	*3152.66*	*8650.76*	*8715.53*
5 (transfer)	*1243.30*	*2.59*	*7192.64*	*1805.36*	*10243.89*	*10340.73*
5(t-outer)	*318.58*	*0.89*	*696.19*	*713.29*	*1728.95*	*1750.42*
ONE METER FROM THE 100-TON HI-TRAC						
1	*422.99*	*17.82*	*142.41*	*76.30*	*659.52*	*692.00*
2	*1400.26*†	*41.25*	*11.27*	*93.37*	*1546.14*	*1655.62*
3	*177.50*	*9.93*	*118.30*	*36.64*	*342.37*	*391.80*
3 (temp)	*176.53*	*10.66*	*100.76*	*13.85*	*301.80*	*346.37*
4	*27.82*	*0.45*	*131.25*	*120.44*	*279.96*	*318.53*
5 (transfer)	*552.69*	*0.48*	*2938.22*	*503.83*	*3995.21*	*4034.34*
5(t-outer)	*76.44*	*1.54*	*264.85*	*144.64*	*487.47*	*491.37*

Notes:
- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

† The cobalt activation of incore grid spacers accounts for *approximately* 6~~.3~~% of the surface and one-meter dose rates.

Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRACS FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
75,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 125-TON HI-TRACs						
1	6.32	61.85	100.63	415.90	584.70	585.42
2	113.33†	183.20	0.01	287.94	584.49	600.36
3	1.41	6.55	62.26	663.65	733.88	753.59
4	41.57	8.40	340.67	767.94	1158.58	1274.01
4 (outer)	4.84	6.00	42.31	16.11	69.26	83.45
5 (pool)	54.77	3.67	454.56	2883.53	3396.53	3404.24
5 (transfer)	65.81	4.78	601.40	440.29	1112.28	1117.76
ONE METER FROM THE 125-TON HI-TRACs						
1	14.93	24.68	12.90	68.44	120.95	122.99
2	50.47†	59.39	0.52	98.23	208.61	215.68
3	5.66	13.95	12.58	61.07	93.26	98.17
4	11.54	2.03	82.02	79.09	174.68	202.33
5 (transfer)	25.98	0.92	290.76	76.26	393.92	396.85

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

† The cobalt activation of incore grid spacers accounts for 9.4% of the surface and one-meter dose rates.

Table 5.1.9

DOSE RATES FOR ARRAYS OF MPC-24
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES

Array Configuration	1 cask	2x2	2x3	2x4	2x5
~~HI-STORM 100 Overpack~~					
~~47,500 MWD/MTU AND 3-YEAR COOLING~~					
~~Annual Dose (mrem/year)~~	~~24.10~~	~~18.07~~	~~15.86~~	~~21.15~~	~~16.29~~
~~Distance to Controlled Area Boundary (meters)~~	~~250~~	~~350~~	~~400~~	~~400~~	~~450~~
~~52,500 MWD/MTU AND 5-YEAR COOLING~~					
~~Annual Dose (mrem/year)†~~	~~22.88~~	~~14.34~~	~~21.52~~	~~16.79~~	~~20.99~~
~~Distance to Controlled Area Boundary (meters)††~~	~~200~~	~~300~~	~~300~~	~~350~~	~~350~~
~~45,000 MWD/MTU AND 9-YEAR COOLING~~					
~~Annual Dose (mrem/year)†~~	~~22.20~~	~~23.41~~	~~16.77~~	~~22.36~~	~~14.91~~
~~Distance to Controlled Area Boundary (meters)††~~	~~150~~	~~200~~	~~250~~	~~250~~	~~300~~
HI-STORM 100S Version B Overpack					
~~47,500~~*60,000* MWD/MTU AND 3-YEAR COOLING					
Annual Dose (mrem/year) †	*19.26*	*16.41*	*24.62*	*20.36*	*16.34*
Distance to Controlled Area Boundary (meters) ††,†††	*350*	*450*	*450*	*500*	*550*
~~52,500 MWD/MTU AND 5-YEAR COOLING~~					
~~Annual Dose (mrem/year)†~~	~~24.86~~	~~15.26~~	~~22.88~~	~~17.24~~	~~21.55~~
~~Distance to Controlled Area Boundary (meters)††~~	~~200~~	~~300~~	~~300~~	~~350~~	~~350~~
45,000 MWD/MTU AND 9-YEAR COOLING					
Annual Dose (mrem/year) †	*23.56*	*14.30*	*21.46*	*15.89*	*19.86*
Distance to Controlled Area Boundary (meters) ††	*200*	*300*	*300*	*350*	*350*

† 8760 hr. annual occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 3-year cooling, as specified in the Section 2.1.9, is lower than the burnup used for this analysis.

Table 5.1.10

DOSE RATES AT ONE METER FROM HI-TRAC
FOR ACCIDENT CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
AT BOUNDING BURNUP AND COOLING TIMES

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
125-TON HI-TRACs					
75,000 MWD/MTU AND 5-YEAR COOLING					
2 (Accident Condition)	92.26	1.02	3476.98	3570.26	3583.16
2 (Normal Condition)	109.86	0.52	98.23	208.61	215.68
100-TON HI-TRAC					
75,000 MWD/MTU AND 5-YEAR COOLING					
2 (Accident Condition)	1354.67	17.88	4359.16	5731.72	5927.95
2 (Normal Condition)	829.09	9.90	168.82	1007.81	1117.29

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.11

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~35~~*45*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
1	*41.37*	*70.98*	*14.80*	*127.15*	*130.10*
2	*239.51*	*0.32*	*4.24*	*244.08*	*261.07*
3	*11.22*	*17.82*	*5.51*	*34.54*	*40.95*
4	*12.02*	*4.29*	*4.11*	*20.43*	*22.78*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.12

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~47,500~~*60,000* MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
1	*34.25*	*57.09*	*29.86*	*121.20*	*122.67*
2	*252.16*	*0.10*	*7.16*	*259.41*	*273.60*
3	*13.56*	*15.57*	*9.82*	*38.94*	*43.90*
4	*13.42*	*4.65*	*7.22*	*25.29*	*27.30*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.13

DOSE RATES ADJACENT TO HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~40~~*50*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*20.15*	*56.22*	*18.80*	*95.17*
2	*211.31*	*0.12*	*6.38*	*217.81*
3	*4.39*	*18.15*	*3.73*	*26.27*
4	*7.76*	*5.05*	*3.40*	*16.20*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.14

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~35~~*45*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point[†] Location	**Fuel Gammas[††] (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
1	*32.51*	*18.19*	*2.41*	*53.10*	*55.78*
2	*124.98*	*1.42*	*1.75*	*128.15*	*136.88*
3	*14.74*	*8.67*	*0.75*	*24.16*	*28.13*
4	*2.79*	*1.31*	*1.16*	*5.26*	*5.89*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.15

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~47,500~~*60,000* MWD/MTU AND 3-YEAR COOLING

Dose Point[†] Location	Fuel Gammas[††] (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*33.12*	*16.07*	*4.80*	*54.00*	*55.76*
2	*129.84*	*1.15*	*2.84*	*133.83*	*141.21*
3	*15.89*	*7.23*	*1.30*	*24.42*	*27.44*
4	*3.22*	*1.41*	*2.36*	*6.99*	*7.54*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.16

DOSE RATES AT ONE METER FROM HI-STORM 100S VERSION B OVERPACK
FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~40~~*50*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*25.76*	*15.91*	*3.21*	*44.88*
2	*107.43*	*0.95*	*2.46*	*110.84*
3	*7.78*	*8.96*	*0.73*	*17.47*
4	*1.78*	*1.65*	*0.84*	*4.27*

† Refer to Figure 5.1.13.

†† Gammas generated by neutron capture are included with fuel gammas.

FIGURE 5.1.1

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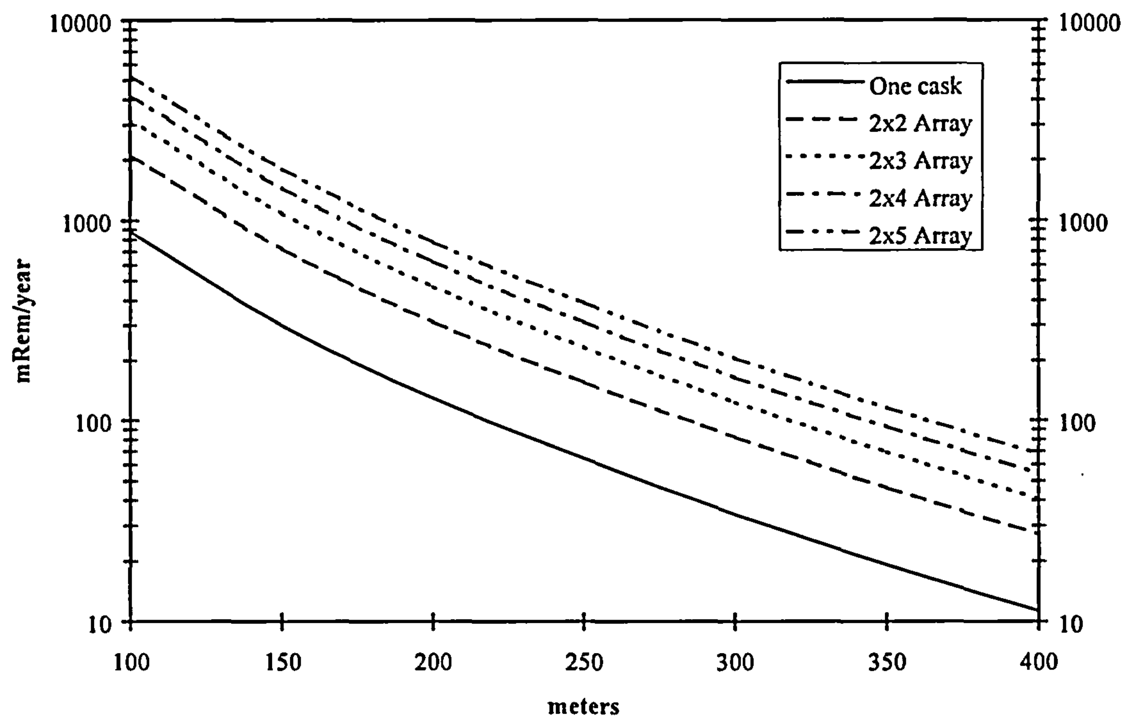

FIGURE 5.1.3; ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 60,000 MWD/MTU AND 3-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

FIGURE 5.1.12

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5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

Sample input files for SAS2H and ORIGEN-S are provided in Appendices 5.A and 5.B, respectively. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of the steel structural material in the fuel element above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. The PWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load for a given burnup and cooling time from the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun. The BWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load for a given burnup and cooling time from the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest UO_2 mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly is described in Table 5.2.2. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.21 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6 and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger ^{60}Co activity.

The design basis stainless steel clad fuel assembly for the Indian Point 1, Haddam Neck, and San Onofre 1 assembly classes is described in Table 5.2.3. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

The design basis assemblies mentioned above are the design basis assemblies for both intact and damaged fuel and fuel debris for their respective array classes. Analyses of damaged fuel is presented in Section 5.4.2.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

5.2.1 Gamma Source

Tables 5.2.4 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuels at varying burnups and cooling times. Tables 5.2.7 and 5.2.22 provides the gamma source in MeV/s and photons/s for the design basis 6x6 and MOX fuel, respectively.

Specific analysis for the HI-STORM 100 System, which includes the HI-STORM storage overpacks and the HI-TRAC transfer casks, was performed to determine the dose contribution from gammas as a function of energy. This analysis considered dose locations external to the 100-ton HI-TRAC transfer cask and the HI-STORM 100 overpack and vents. The results of this analysis have revealed that, due to the magnitude of the gamma source at lower energies, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose at all high dose locations). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Dose rate contributions from above and below this range were evaluated and found to be negligible. Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Conservatively, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

Holtec International has gathered information from utilities and vendors which shows that the 1.0 gm/kg impurity level is very conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. The typical Cobalt-59 impurity level for fuel since the late 1980s is less than 0.5 gm/kg. Based on this, fuel with a short cooling time, 5 to 9 years, would have a Cobalt-59 impurity level less than 0.5 gm/kg. Therefore, the use of a bounding Cobalt-59 impurity level of 1.0 gm/kg is very conservative, particularly for recently manufactured assemblies. Analysis in Reference [5.2.3] indicates that the cobalt impurity in steel and inconel for fuel manufactured in the 1970s ranged from approximately 0.2 gm/kg to 2.2 gm/kg. However, older fuel manufactured with higher cobalt impurity levels will also have a corresponding longer cooling time and therefore will be bounded by the analysis presented in this chapter. As confirmation of this statement, Appendix D presents a comparison of the dose rates around the 100-ton HI-TRAC and the HI-STORM with the MPC-24 for a short cooling time (5 years) using the 1.0 gm/kg mentioned above and for a long cooling time (9 years) using a higher cobalt impurity level of 4.7 gm/kg for inconel. These results confirm that the dose rates for the longer cooling time with the higher impurity level are essentially equivalent to (within 11%) or bounded by the dose rates for the shorter cooling time with the lower impurity level. Therefore, the analysis in this chapter is conservative.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM 100 system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 1 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. The masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ^{59}Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

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

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.10. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.11 through 5.2.13 provide the ^{60}Co activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the MPC-32, MPC-24, and the MPC-68 for varying burnup and cooling times. The design basis 6x6 and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR design basis fuel. This is a conservative assumption as the design basis 6x6 fuel and MOX fuel assemblies are limited to a significantly lower burnup and longer cooling time than the design basis fuel.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cs-134 and Eu-154, two of the major contributors to the gamma source, range from 0.79 to 1.009 and 0.79 to 0.98, respectively. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - 75,000 MWD/MTU for PWR and 20,000 - 70,000 MWD/MTU for BWR zircaloy clad fuel. These enrichments are based on References [5.2.6] and [5.2.7]. Table 8 of reference [5.2.6] presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, for burnups up to 50,000 MWD/MTU, are approximately the average enrichments from Table 8 of reference [5.2.6] for the burnup range that is 5,000 MWD/MTU less than the ranges listed in

Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] and some additional data from nuclear plants was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the shielding analysis are adequate to bound the fuel authorized by the limits in Section 2.1.9 for loading in the HI-STORM system. Since the enrichment does affect the source term evaluation, it is recommended that the site-specific dose evaluation consider the enrichment for the fuel being stored.

The neutron source calculated for the design basis fuel assemblies for the MPC-24, MPC-32, and MPC-68 and the design basis 6x6 fuel are listed in Tables 5.2.15 through 5.2.18 in neutrons/s for varying burnup and cooling times. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately 92-97% of the total number of neutrons produced. Alpha,n reactions in isotopes other than ^{244}Cm account for approximately 0.3-2% of the neutrons produced while spontaneous fission in isotopes other than ^{244}Cm account for approximately 2-8% of the neutrons produced within the UO_2 fuel. In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cm-244 ranges from 0.81 to 0.95. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.3 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.3 is actually longer than the true active fuel length of 122 inches for the WE 15x15 and 83 inches for the LaCrosse 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this does not reflect the potential change in dose rates. As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 neutrons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches and 144 inches, respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 neutrons/s/inch and 1 neutron/s/inch for the stainless steel and zircaloy fuel, respectively. The result would be a higher neutron dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.8, 5.2.9, 5.2.19, and 5.2.20 list the gamma and neutron source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV, which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.4 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

5.2.4 Non-fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM 100 System as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted to the inner four fuel storage locations in the MPC-24, MPC-24E, and the MPC-32.

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding

BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable Co-60 source and decay heat from the BPRA and TPD were specified as: 50 curies Co-60 and 0.77 watts for each TPD and 895 curies Co-60 and 14.4 watts for each BPRA. Table 5.2.31 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Section 5.4.6 discusses the increase in the cask dose rates due to the insertion of BPRAs or TPDs into fuel assemblies.

5.2.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W

stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of ~~CRAs and~~ APSRs can vary, the quantity that can be stored in an MPC is being limited. *Four APSRs are permitted in the MPC-24 and twelve APSRs are permitted in the MPC-32.* ~~to four CRAs and/or APSRs~~. These ~~four~~ devices are required to be stored in the ~~inner four~~ locations ~~in the MPC-24, MPC-24E, MPC-24EF, and MPC-32 as~~ outlined in Section 2.1.9.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

<u>Configuration 1: CRA and APSR</u>
This configuration had the lower 15 inches of the CRA and APSR activated at full flux with two regions above the 15 inches activated at a reduced power level. This simulates a CRA or APSR which was operated at 10% insertion. The regions above the 15 inches reflect the upper portion of the fuel assembly.

<u>Configuration 2: CRA and APSR</u>
This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

Configuration 3: APSR

This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.

Tables 5.2.34 and 5.2.35 present the source terms, including decay heat, that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in Section 2.1.9. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Section 5.4.6 discusses the effect on dose rate of the insertion of APSRs and CRAs into the inner four fuel assemblies in the MPC-24 or MPC-32.

5.2.5 Choice of Design Basis Assembly

The analysis presented in this chapter was performed to bound the fuel assembly classes listed in Tables 2.1.1 and 2.1.2. In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly that produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the PWR MPCs (MPC-24 and MPC-32) and the BWR MPCs (MPC-68).

5.2.5.1 PWR Design Basis Assembly

Table 2.1.1 lists the PWR fuel assembly classes that were evaluated to determine the design basis PWR fuel assembly. Within each class, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within a class are very minor (pellet diameter, clad thickness, etc.), it is conservative to choose the assembly with the highest UO_2 mass. For a given class 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, the highest UO_2 mass will have produced the most energy and therefore the most fission products.

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table. The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Indian Point 1, Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO_2 mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. The power/assembly values used in Table 5.2.25 were calculated by dividing 110% of the thermal power for commercial PWR reactors using that array class by the number of assemblies in the core. The higher thermal power, 110%, was used to account for potential power uprates. The power level used for the B&W15 is an additional 17% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

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

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

5.2.5.2 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. ~~A penalty was not applied to the PWR array classes since the thermal analysis in Chapter 4 has more than a 5% margin in the calculated allowable decay heat.~~

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.

5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STORM 100 System.

5.2.7 Fuel Assembly Neutron Sources

Neutron sources are used in reactors during initial startup of reactor cores. There a different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STORM 100 System. Currently these are the only neutron source permitted for storage in the HI-STORM 100 System.

5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which, have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and Section 2.1.9 requires that these channels be stored in the inner sixteen locations.

The activation of a single stainless steel channel was calculated by simulating the irradiation of the channels with ORIGEN-S using the flux calculated from the LaCrosse fuel assembly. The mass of the steel channel in the active fuel zone (83 inches) was used in the analysis. For burnups beyond 22,500 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 22,500 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 22,500 MWD/MTU.

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming the plant operated continually from 11/69 until 4/87, approximately 17.5 years or 6388 days, the accumulated burnup for the channels would be 186,000 MWD/MTU (6388 days times 29.17 MW/MTU from Table 5.2.3). Therefore, the cobalt activity calculated for a single stainless steel channel irradiated for 180,000 MWD/MTU was calculated to be 667 curies of Co-60 for 13 years cooling. This is equivalent to a source of 4.94E+13 photons/sec in the energy range of 1.0-1.5 MeV.

In order to demonstrate that sixteen stainless steel channels are acceptable for storage in an MPC-68, a comparison of source terms is performed. Table 5.2.8 indicates that the source term for the LaCrosse design basis fuel assembly in the 1.0-1.5 MeV range is 6.34E+13 photons/sec for 10 years cooling, assuming a 144 inch active fuel length. This is equivalent to 4.31E+15 photons/sec/cask. At 13 years cooling, the fuel source term in that energy range decreases to 4.31E+13 photons/sec which is equivalent to 2.93E+15 photons/sec/cask. If the source term from the stainless steel channels is scaled to 144 inches and added to the 13 year fuel source term the result is 4.30E+15 photons/sec/cask (2.93E+15 photons/sec/cask + 4.94E+13 photons/sec/channel x 144 inch/83 inch x 16 channels/cask). This number is equivalent to the 10 year 4.31E+15 photons/sec/cask source calculated from Table 5.2.8 and used in the shielding analysis in this chapter. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	PWR	**BWR**
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)[††]	562.029	225.177
Weight of U (kg)[††]	495.485	198.516

Notes:

1. The B&W 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun.
2. The GE 7x7 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR	BWR
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2

DESCRIPTION OF DESIGN BASIS GE 6x6 ZIRCALOY CLAD FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2 (kg)†	129.5
Weight of U (kg)†	114.2

Notes:
1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

† Derived from parameters in this table.

Table 5.2.3

DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

	PWR	BWR
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.5	3.5
Burnup (MWD/MTU)†	40,000 (MPC-24 and 32)	22,500 (MPC-68)
Cooling Time (years)†	8 (MPC-24), 9 (MPC-32)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

Notes:
1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: Indian Point 1, Haddam Neck, and San Onofre 1.
2. The LaCrosse 10x10 is the design basis assembly for the following fuel assembly class listed in Table 2.1.2: LaCrosse.

† Burnup and cooling time combinations are equivalent to or conservatively bound the limits in Section 2.1.9.

Table 5.2.4

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**~~35~~*45*,000 MWD/MTU 3 Year Cooling**		**~~75~~*69*,000 MWD/MTU ~~8~~-*5* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*3.05E+15*	*5.30E+15*	*3.26E+15*	*5.67E+15*
0.7	1.0	*1.37E+15*	*1.62E+15*	*1.23E+15*	*1.44E+15*
1.0	1.5	*2.96E+14*	*2.37E+14*	*2.69E+14*	*2.15E+14*
1.5	2.0	*2.91E+13*	*1.66E+13*	*1.41E+13*	*8.08E+12*
2.0	2.5	*3.79E+13*	*1.68E+13*	*7.56E+12*	*3.36E+12*
2.5	3.0	*1.14E+12*	*4.13E+11*	*3.56E+11*	*1.29E+11*
Total		*4.78E+15*	*7.18E+15*	*4.78E+15*	*7.34E+15*

Table 5.2.5

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**~~46~~*60*,000 MWD/MTU 3 Year Cooling**		**~~47,500 MWD/MTU 3 Year Cooling~~**		**75,000 MWD/MTU 5 Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	~~(MeV/s)~~	~~(Photons/s)~~	(MeV/s)	(Photons/s)
0.45	0.7	*4.11E+15*	*7.14E+15*	~~3.25E+15~~	~~5.65E+15~~	3.55E+15	6.17E+15
0.7	1.0	*1.98E+15*	*2.33E+15*	~~1.49E+15~~	~~1.75E+15~~	1.36E+15	1.60E+15
1.0	1.5	*4.04E+14*	*3.23E+14*	~~3.17E+14~~	~~2.53E+14~~	2.94E+14	2.35E+14
1.5	2.0	*3.41E+13*	*1.95E+13*	~~3.03E+13~~	~~1.73E+13~~	1.50E+13	8.59E+12
2.0	2.5	*3.95E+13*	*1.76E+13*	~~3.83E+13~~	~~1.70E+13~~	7.63E+12	3.39E+12
2.5	3.0	*1.29E+12*	*4.70E+11*	~~1.19E+12~~	~~4.33E+11~~	3.72E+11	1.35E+11
Total		*6.57E+15*	*9.84E+15*	~~5.12E+15~~	~~7.69E+15~~	5.23E+15	8.02E+15

Table 5.2.6

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	Upper Energy	~~39~~*50*,000 MWD/MTU 3 Year Cooling		~~40,000 MWD/MTU 3 Year Cooling~~		~~70,000 MWD/MTU 6 Year Cooling~~	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	~~(MeV/s)~~	~~(Photons/s)~~	~~(MeV/s)~~	~~(Photons/s)~~
0.45	0.7	*1.28E+15*	*2.23E+15*	~~1.02E+15~~	~~1.78E+15~~	~~1.10E+15~~	~~1.91E+15~~
0.7	1.0	*5.76E+14*	*6.77E+14*	~~4.37E+14~~	~~5.14E+14~~	~~3.21E+14~~	~~3.78E+14~~
1.0	1.5	*1.18E+14*	*9.47E+13*	~~9.40E+13~~	~~7.52E+13~~	~~7.67E+13~~	~~6.13E+13~~
1.5	2.0	*1.04E+13*	*5.92E+12*	~~9.27E+12~~	~~5.30E+12~~	~~3.55E+12~~	~~2.03E+12~~
2.0	2.5	*1.20E+13*	*5.33E+12*	~~1.17E+13~~	~~5.21E+12~~	~~1.03E+12~~	~~4.57E+11~~
2.5	3.0	*4.04E+11*	*1.47E+11*	~~3.70E+11~~	~~1.35E+11~~	~~5.83E+10~~	~~2.12E+10~~
Total		*2.00E+15*	*3.01E+15*	~~1.58E+15~~	~~2.38E+15~~	~~1.50E+15~~	~~2.35E+15~~

Table 5.2.7

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.53e+14	2.65e+14
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+09	5.99e+08
2.5	3.0	7.30e+07	2.66e+07
Totals		1.61e+14	2.73e+14

Table 5.2.8

CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	**Upper Energy**	**22,500 MWD/MTU 10-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	2.72e+14	4.74+14
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		3.72e+14	5.61e+14

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 83-inch active fuel length.

Table 5.2.9

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 8-Year Cooling**		**40,000 MWD/MTU 9-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.37e+15	2.38e+15	1.28E+15	2.22E+15
7.0e-01	1.0	2.47e+14	2.91e+14	1.86E+14	2.19E+14
1.0	1.5	4.59e+14	3.67e+14	4.02E+14	3.21E+14
1.5	2.0	3.99e+12	2.28e+12	3.46E+12	1.98E+12
2.0	2.5	5.85e+11	2.60e+11	2.69E+11	1.20E+11
2.5	3.0	3.44e+10	1.25e+10	1.77E+10	6.44E+09
Totals		2.08e+15	3.04e+15	1.87E+15	2.76E+15

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 122-inch active fuel length.

Table 5.2.10

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

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

Table 5.2.11

CALCULATED MPC-32 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
ZIRCALOY CLAD FUEL
AT DESIGN BASIS BURNUP AND COOLING TIME

Location	~~35~~*45*,000 MWD/MTU and 3-Year Cooling (curies)	~~75~~*69*,000 MWD/MTU and *8*5-Year Cooling (curies)
Lower End Fitting	*217.58*	*208.12*
Gas Plenum Springs	*16.60*	*15.88*
Gas Plenum Spacer	*9.52*	*9.11*
Expansion Springs	N/A	N/A
Incore Grid Spacers	*563.50*	*539.00*
Upper End Fitting	*106.72*	*102.08*
Handle	N/A	N/A

Table 5.2.12

CALCULATED MPC-24 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
ZIRCALOY CLAD FUEL
AT DESIGN BASIS BURNUP AND COOLING TIME

Location	**~~46~~*60*,000 MWD/MTU and 3-Year Cooling (curies)**	**~~47,500 MWD/MTU and 3-Year Cooling (curies)~~**	**75,000 MWD/MTU and - 5 Year Cooling (curies)**
Lower End Fitting	*249.74*	~~227.04~~	219.47
Gas Plenum Springs	*19.05*	~~17.32~~	16.74
Gas Plenum Spacer	*10.93*	~~9.94~~	9.61
Expansion Springs	N/A	~~N/A~~	N/A
Incore Grid Spacers	*646.80*	~~588.00~~	568.40
Upper End Fitting	*122.50*	~~111.36~~	107.65
Handle	N/A	~~N/A~~	N/A

Table 5.2.13

CALCULATED MPC-68 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL AT DESIGN BASIS BURNUP AND COOLING TIME

Location	**~~39~~*50*,000 MWD/MTU and 3-Year Cooling (curies)**	**~~40,000 MWD/MTU and 3-Year Cooling (curies)~~**	**~~70,000 MWD/MTU and 6-Year Cooling (curies)~~**
Lower End Fitting	*90.55*	~~82.69~~	~~68.73~~
Gas Plenum Springs	*27.67*	~~25.27~~	~~21.00~~
Gas Plenum Spacer	N/A	~~N/A~~	~~N/A~~
Expansion Springs	*5.03*	~~4.59~~	~~3.82~~
Grid Spacer Springs	*41.50*	~~37.90~~	~~31.50~~
Upper End Fitting	*25.15*	~~22.97~~	~~19.09~~
Handle	*3.14*	~~2.87~~	~~2.39~~

Table 5.2.14

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Table 5.2.15

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	~~35~~*45*,000 MWD/MTU 3-Year Cooling (Neutrons/s)	~~75~~*69*,000 MWD/MTU 8*5*-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	*1.77E+07*	*5.31E+07*
4.0e-01	9.0e-01	*9.03E+07*	*2.71E+08*
9.0e-01	1.4	*8.27E+07*	*2.48E+08*
1.4	1.85	*6.09E+07*	*1.82E+08*
1.85	3.0	*1.08E+08*	*3.21E+08*
3.0	6.43	*9.77E+07*	*2.92E+08*
6.43	20.0	*8.66E+06*	*2.60E+07*
Totals		*4.65E+08*	*1.39E+09*

Table 5.2.16

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**~~46~~*60*,000 MWD/MTU 3-Year Cooling (Neutrons/s)**	**~~47,500 MWD/MTU 3-Year Cooling (Neutrons/s)~~**	**75,000 MWD/MTU 5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*3.76E+07*	~~2.19E+07~~	6.82E+07
4.0e-01	9.0e-01	*1.92E+08*	~~1.12E+08~~	3.48E+08
9.0e-01	1.4	*1.76E+08*	~~1.02E+08~~	3.18E+08
1.4	1.85	*1.29E+08*	~~7.54E+07~~	2.34E+08
1.85	3.0	*2.28E+08*	~~1.33E+08~~	4.11E+08
3.0	6.43	*2.08E+08*	~~1.21E+08~~	3.75E+08
6.43	20.0	*1.84E+07*	~~1.07E+07~~	3.34E+07
Totals		*9.89E+08*	~~5.76E+08~~	1.79E+09

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**~~39~~*50*,000 MWD/MTU 3-Year Cooling (Neutrons/s)**	~~**40,000 MWD/MTU 3-Year Cooling (Neutrons/s)**~~	~~**70,000 MWD/MTU 6-Year Cooling (Neutrons/s)**~~
1.0e-01	4.0e-01	*9.79E+06*	~~5.45E+06~~	~~1.98E+07~~
4.0e-01	9.0e-01	*5.00E+07*	~~2.78E+07~~	~~1.01E+08~~
9.0e-01	1.4	*4.57E+07*	~~2.55E+07~~	~~9.26E+07~~
1.4	1.85	*3.37E+07*	~~1.88E+07~~	~~6.81E+07~~
1.85	3.0	*5.93E+07*	~~3.32E+07~~	~~1.20E+08~~
3.0	6.43	*5.40E+07*	~~3.02E+07~~	~~1.09E+08~~
6.43	20.0	*4.79E+06*	~~2.67E+06~~	~~9.71E+06~~
Totals		*2.57E+08*	~~1.44E+08~~	~~5.20E+08~~

Table 5.2.18

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Total		2.20e+7

Table 5.2.19

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**22,500 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 83-inch active fuel length.

Table 5.2.20

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 8-Year Cooling (Neutrons/s)	40,000 MWD/MTU 9-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.04e+7	1.01E+07
4.0e-01	9.0e-01	5.33e+7	5.14E+07
9.0e-01	1.4	4.89e+7	4.71E+07
1.4	1.85	3.61e+7	3.48E+07
1.85	3.0	6.41e+7	6.18E+07
3.0	6.43	5.79e+7	5.58E+07
6.43	20.0	5.11e+6	4.92E+06
Totals		2.76e+8	2.66E+08

Note: These source terms were calculated for a 144-inch fuel length. The limits in Section 2.1.9 are based on the actual 122-inch active fuel length.

Table 5.2.21

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO_2 and $PuUO_2$
No. of UO_2 Rods	27
No. of $PuUO_2$ rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)†	2.24 (UO_2 rods) 0.711 ($PuUO_2$ rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2,$PuUO_2$ (kg)††	123.3
Weight of U,Pu (kg)††	108.7

† See Table 5.3.3 for detailed composition of $PuUO_2$ rods.

†† Derived from parameters in this table.

Table 5.2.22

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.45e+14	2.52e+14
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		1.53e+14	2.60e+14

Table 5.2.23

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.24

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	**Initial Enrichment (wt.% ^{235}U)**
BWR Fuel	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	3.6
55,000-60,000	4.0
60,000-65,000	4.4
65,000-70,000	4.8
PWR Fuel	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	3.9
55,000-60,000	4.2
60,000-65,000	4.5
65,000-70,000	4.8
70,000-75,000	5.0

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. This note does not apply to the maximum burnups of 70,000 and 75,000 MWD/MTU.

Table 5.2.25 (page 1 of 2)

DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	WE 14×14	WE 14x14	WE 15×15	WE 17×17	WE 17x17
Fuel assembly array class	14x14B	14x14A	15x15AB C	17x17B	17x17A
Active fuel length (in.)	144	144	144	144	144
No. of fuel rods	179	179	204	264	264
Rod pitch (in.)	0.556	0.556	0.563	0.496	0.496
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.422	0.4	0.422	0.374	0.36
Cladding thickness (in.)	0.0243	0.0243	0.0245	0.0225	0.0225
Pellet diameter (in.)	0.3659	0.3444	0.3671	0.3232	0.3088
Pellet material	UO_2	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (% of theoretical)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% ^{235}U)	3.4	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	15.0	15.0	18.6	20.4	20.4
Specific power (MW/MTU)	36.409	41.097	39.356	43.031	47.137
Weight of UO_2 (kg)†	467.319	414.014	536.086	537.752	490.901
Weight of U (kg)†	411.988	364.994	472.613	474.082	432.778
No. of Guide Tubes	17	17	21	25	25
Guide Tube O.D. (in.)	0.539	0.539	0.546	0.474	0.474
Guide Tube Thickness (in.)	0.0170	0.0170	0.0170	0.0160	0.0160

† Derived from parameters in this table.

Table 5.2.25 (page 2 of 2)

DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Fuel assembly array class	14x14C	16x16A	15x15DEF H	17x17C
Active fuel length (in.)	144	150	144	144
No. of fuel rods	176	236	208	264
Rod pitch (in.)	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	0.3805	0.3255	0.3742	0.3252
Pellet material	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	10.522 (96%)	10.522 (96%)	10.412 (95%)	10.522 (96%)
Enrichment (wt.% ^{235}U)	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Power/assembly (MW)	13.7	17.5	19.819	20.4
Specific power (MW/MTU)	31.275	39.083	40	42.503
Weight of UO_2 (kg)†	496.887	507.9	562.029	544.428
Weight of U (kg)†	438.055	447.764	495.485	479.968
No. of Guide Tubes	5	5	17	25
Guide Tube O.D. (in.)	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0400	0.0400	0.0160	0.0175

† Derived from parameters in this table.

Table 5.2.26 (page 1 of 2)

DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	7×7	8×8	8x8	9x9	9x9
Fuel assembly array class	7x7B	8x8B	8x8CDE	9x9A	9x9B
Active fuel length (in.)	144	144	150	144	150
No. of fuel rods	49	64	62	74	72
Rod pitch (in.)	0.738	0.642	0.64	0.566	0.572
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.484	0.493	0.44	0.433
Cladding thickness (in.)	0.0355	0.02725	0.034	0.028	0.026
Pellet diameter (in.)	0.488	0.4195	0.416	0.376	0.374
Pellet material	UO_2	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (% of theoretical)	10.412 (95%)	10.412 (95%)	10.412 (95%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% ^{235}U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	5.96	5.75	5.75	5.75	5.75
Specific power (MW/MTU)	30	30	30.24	31.97	31.88
Weight of UO_2 (kg)†	225.177	217.336	215.673	204.006	204.569
Weight of U (kg)†	198.516	191.603	190.137	179.852	180.348
No. of Water Rods	0	0	2	2	1
Water Rod O.D. (in.)	n/a	n/a	0.493	0.98	1.516
Water Rod Thickness (in.)	n/a	n/a	0.034	0.03	0.0285

† Derived from parameters in this table.

Table 5.2.26 (page 1 of 2)

DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	9x9	9×9	9x9	10×10	10x10
Fuel assembly array class	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Active fuel length (in.)	150	144	150	144	150
No. of fuel rods	80	76	72	92	96
Rod pitch (in.)	0.572	0.572	0.572	0.510	0.488
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.423	0.443	0.424	0.404	0.378
Cladding thickness (in.)	0.0295	0.0285	0.03	0.0260	0.0243
Pellet diameter (in.)	0.3565	0.3745	0.3565	0.345	0.3224
Pellet material	UO_2	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (% of theoretical)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)	10.522 (96%)
Enrichment (wt.% ^{235}U)	3.0	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5
Power/assembly (MW)	5.75	5.75	5.75	5.75	5.75
Specific power (MW/MTU)	31.58	31.38	35.09	30.54	32.18
Weight of UO_2 (kg)†	206.525	207.851	185.873	213.531	202.687
Weight of U (kg)†	182.073	183.242	163.865	188.249	178.689
No. of Water Rods	1	5	1	2	1
Water Rod O.D. (in.)	0.512	0.546	1.668	0.980	Note 1
Water Rod Thickness (in.)	0.02	0.0120	0.032	0.0300	Note 1

Note 1: 10x10C has a diamond shaped water rod with 4 additional segments dividing the fuel rods into four quadrants.

† Derived from parameters in this table.

Table 5.2.27

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD PWR FUEL
3.4 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly	WE 14×14	WE 14×14	WE 15×15	WE 17×17	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Array class	14x14A	14x14B	15x15 ABC	17x17A	17x17B	14x14C	16x16A	15x15 DEFH	17x17C
Neutrons/sec	1.76E+8 1.78E+8	2.32E+8 2.35E+8	2.70E+8 2.73E+8	2.18E+8	2.68E+8	2.32E+8	2.38E+8	2.94E+8	2.68E+8
Photons/sec (0.45-3.0 MeV)	2.88E+15 2.93E+15	3.28E+15 3.32E+15	3.80E+15 3.86E+15	3.49E+15	3.85E+15	3.37E+15	3.57E+15	4.01E+15	3.89E+15
Thermal power (watts)	809.5 820.7	923.5933. 7	10731086	985.6	1090	946.6	1005	1137	1098

Note:
The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.28

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD BWR FUEL
3.0 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly	7x7	8x8	8x8	9x9	9x9	9x9	9x9	9x9	10x10	10x10
Array Class	7x7B	8x8B	8x8CDE	9x9A	9x9B	9x9CD	9x9EF	9x9G	10x10AB	10x10C
Neutrons/sec	1.33E+8	1.22E+8	1.22E+8	1.13E+8	1.06E+8	1.09E+8	1.24E+8	9.15E+7	1.24E+8	1.07E+8
Photons/sec (0.45-3.0 MeV)	1.55E+15	1.49E+15	1.48E+15	1.41E+15	1.40E+15	1.42E+15	1.45E+15	1.28E+15	1.48E+15	1.40E+15
Thermal power (watts)	435.5	417.3	414.2	394.2	389.8	395	405.8	356.9	413.5	389.2

Table 5.2.29

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL AND VALUES REPORTED IN THE DOE CHARACTERISTICS DATABASE† FOR 30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Source Term Calculations (watts/assembly)
PWR Fuel		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	802.7
CE 16x16	653.7	734.3
CE 14x14	601.3	694.9
WE 17x17	742.5	795.4
WE 15x15	762.2	796.2
WE 14x14	649.6	682.9
BWR Fuel		
7x7	310.9	315.7
8x8	296.6	302.8
9x9	275.0	286.8

Notes:

1. The decay heat from the source term calculations is the maximum value calculated for that fuel assembly class.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.
4. The enrichments used for the column labeled "Decay Heat from Source Term Calculations" were consistent with Table 5.2.24.

† Reference [5.2.7].

Table 5.2.30

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

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

Table 5.2.31

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES

Region	BPRA	TPD
Upper End Fitting (curies Co-60)	32.7	25.21
Gas Plenum Spacer (curies Co-60)	5.0	9.04
Gas Plenum Springs (curies Co-60)	8.9	15.75
In-core (curies Co-60)	848.4	N/A

Table 5.2.32

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			**Flux Weighting Factor**	**Mass of cladding (kg Inconel)**	**Mass of absorber (kg AgInCd)**
Start (in)	**Finish (in)**	**Length (in)**			
Configuration 1 - 10% Inserted					
0.0	15.0	15.0	1.0	1.32	7.27
15.0	18.8125	3.8125	0.2	0.34	1.85
18.8125	28.25	9.4375	0.1	0.83	4.57
Configuration 2 - Fully Removed					
0.0	3.8125	3.8125	0.2	0.34	1.85
3.8125	13.25	9.4375	0.1	0.83	4.57

Table 5.2.33

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD CONFIGURATION S FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Steel)	Mass of absorber (kg Inconel)
Start (in)	Finish (in)	Length (in)			
Configuration 1 - 10% Inserted					
0.0	15.0	15.0	1.0	1.26	5.93
15.0	18.8125	3.8125	0.2	0.32	1.51
18.8125	28.25	9.4375	0.1	0.79	3.73
Configuration 2 - Fully Removed					
0.0	3.8125	3.8125	0.2	0.32	1.51
3.8125	13.25	9.4375	0.1	0.79	3.73
Configuration 3 - Fully Inserted					
0.0	63.0	63.0	1.0	5.29	24.89
63.0	66.8125	3.8125	0.2	0.32	1.51
66.8125	76.25	9.4375	0.1	0.79	3.73

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Photons/sec from AgInCd			Curies Co-60 from Inconel
Start (in)	Finish (in)	Length (in)	0.3-0.45 MeV	0.45-0.7 MeV	0.7-1.0 MeV	
Configuration 1 - 10% Inserted - 80.8 watts decay heat						
0.0	15.0	15.0	1.91e+14	1.78e+14	1.42e+14	1111.38
15.0	18.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
18.8125	28.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92
Configuration 2 - Fully Removed - 8.25 watts decay heat						
0.0	3.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
3.8125	13.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER SHAPING ROD CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Curies of Co-60
Start (in)	**Finish (in)**	**Length (in)**	
Configuration 1 - 10% Inserted - 46.2 watts decay heat			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78
Configuration 2 - Fully Removed - 4.72 watts decay heat			
0.0	3.8125	3.8125	136.36
3.8125	13.25	9.4375	168.78
Configuration 3 - Fully Inserted - 178.9 watts decay heat			
0.0	63.0	63.0	11266.80
63.0	66.8125	3.8125	136.36
66.8125	76.25	9.4375	168.78

Table 5.2.36

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

	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
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

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

Table 5.2.38

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA 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

5.3 MODEL SPECIFICATIONS

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

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Section 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

5.3.1 Description of the Radial and Axial Shielding Configuration

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

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. 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 ~~in this chapter~~ were modeled as 1.25 and 0.75 inch thick shells.

Figures 5.3.11, 5.3.18, and 5.3.22 are axial representations of the HI-STORM 100, HI-STORM 100S, and HI-STORM 100S Version B overpacks, respectively, with the various as-modeled dimensions indicated.

Only the HI-STORM 100S Version B is analyzed in this chapter. This is reasonable because the HI-STORM 100S Version B overpack is shorter than the other overpacks, and the MPC is positioned closer to the inlet vent which results in higher dose rates at the inlet vent compared to the other overpacks. In addition, the HI-STORM 100S Version B has slightly higher offsite dose than the other overpacks.

Figures 5.3.12 and 5.3.13 show axial cross-sectional views of the 100- and 125-ton HI-TRAC transfer casks, respectively, with the as-modeled dimensions and materials specified. Figures 5.3.14, 5.3.15, and 5.3.20 show fully labeled radial cross-sectional views of the HI-TRAC 100, 125, and 125D transfer casks, respectively. Finally, Figures 5.3.16 and 5.3.17 show fully labeled diagrams of the transfer lids for the HI-TRAC 100 and 125 transfer casks. Since lead plate may be used instead of poured lead in the pool and transfer lids, there exists the possibility of a gap between the lead plate and the surrounding steel walls. This gap was accounted for in the analysis as depicted on Figures 5.3.16 and 5.3.17. The gap was not modeled in the pool lid since the gap will only exist on the outer edges of the pool lid and the highest dose rate is in the center. (All results presented in this chapter were calculated with the gap with the exception of the results presented in Figures 5.1.6, 5.1.7, and 5.1.11 which did not include the gap.) The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid in this chapter are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. Consistent with the analysis of the transfer lid in which only the portion of the lid directly below the MPC was modeled, the structure of the mating device which surrounds the pool lid was not modeled.

Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. The HI-TRAC 125D has steel ribs in the lower water jacket while the HI-TRAC 125 does not. These additional ribs in the lower water jacket reduce the dose rate in the vicinity of the pool lid for the HI-TRAC 125D compared to the HI-TRAC 125. Since the dose rates at the midplane of the HI-TRAC 125D are higher than the HI-TRAC 125, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.

To reduce the gamma dose around the inlet and outlet vents, stainless steel cross plates, designated gamma shield cross plates[†] (see Figures 5.3.11 and 5.3.18), have been installed inside all vents in all overpacks. The steel in these plates effectively attenuates the fuel and ^{60}Co gammas that dominated the dose at these locations prior to their installation. Figure 5.3.19 shows

[†] This design embodiment, formally referred to as "Duct Photon Attenuator," has been disclosed as an invention by Holtec International for consideration by the US Patent Office for issuance of a patent under U.S. law.

three designs for the gamma shield cross plates to be used in the inlet and outlet vents. The designs in the top portion of the figure are mandatory for use in the HI-STORM 100 and 100S overpacks during normal storage operations and were assumed to be in place in the shielding analysis. The designs in the middle portion of the figure may be used instead of the mandatory designs in the HI-STORM 100S overpack to further reduce the radiation dose rates at the vents. These optional gamma shield cross plates could further reduce the dose rate at the vent openings by as much as a factor of two. The designs in the bottom portion of the figure are mandatory for use in the HI-STORM 100S Version B overpack during normal storage operations and were assumed to be in place in the shielding analysis.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 15 times the pitch and 7 times the pitch, respectively. Homogenization resulted in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The basket material in the top and bottom 0.9 inches where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9-inch area at the top and bottom of the MPC basket.

2. The upper and lower fuel spacers are not modeled, as the fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The fuel spacer length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis, but the fuel spacer materials are not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.

3. For the MPC-32, MPC-24, and MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel that would provide a small increase in shielding. The optional aluminum heat conduction elements are also conservatively not modeled.

4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel that would provide a small amount of additional shielding.

5. In the modeling of the BWR fuel assemblies, the zircaloy flow channels were not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.

6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches compared to 6.25 inches or 7.5 inches.

7. The MPC-68 is designed for two lid thicknesses: 9.5 inches and 10 inches. Conservatively, all calculations reported in this chapter were performed with the 9.5 inch thick lid.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STORM 100 and HI-TRAC. Therefore, the models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in both the HI-TRAC and the HI-STORM outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-TRAC or the HI-STORM.

2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Since the new MPC-24 configuration and the configuration of the MPC-24E are almost identical, the analysis of the earlier MPC-24 configuration is valid for the MPC-24E as well. Figure 5.3.21 shows the superseded and current configuration for the MPC-24 for comparison.

3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches to 0.0235 inches. However, the model still uses 0.06 inches. This discrepancy is compensated for by the use of 9/32 inch cell walls and 5 inch boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration in the 100-ton HI-TRAC for comparison to the superceded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface. These results demonstrate that using the superceded MPC-24 design is conservative.

HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the HI-TRAC 125 was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces effective shielding thickness. The HI-TRAC 125D does not use pocket trunnions.

2. The lifting blocks in the top lid of the 125-ton HI-TRACs were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.

3. The door side plates that are in the middle of the transfer lid of the HI-TRAC 125 are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.

4. The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRACs was modeled as 4 inches larger than it is due to a design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.

HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.

3. In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.

4. The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.

5. The pedestal baseplate, which is steel with holes for pouring concrete, in the HI-STORM 100 and 100S overpacks was modeled as concrete rather than steel. This is acceptable because this piece of steel is positioned at the bottom of the pedestal below 5 inches of steel and a minimum of 11.5 inches of concrete and therefore will have no impact on the dose rates at the bottom vent.

6. Minor penetrations in the body of the overpack (e.g. holes for grounding straps) are not modeled as these are small localized effects which will not affect the off-site dose rates.

7. *Deleted*~~In June 2001, the inner shield shell of the HI-STORM 100 overpack was removed and the concrete density in the body of the overpack (not the pedestal of lid) was increased to compensate. Appendix 5.E presents a comparison of the dose rates calculated for a HI-STORM 100 overpack with and without the inner shield shell. The MPC-24 was used in this comparison. The results indicate that there is very little difference in the calculated dose rates when the inner shield shell is removed and the concrete density is increased. Therefore, all HI-STORM 100 analysis presented in the main portion of this chapter includes the inner shield shell.~~

8. The drawings in Section 1.5 indicate that the HI-STORM 100S has a variable height. This is achieved by adjusting the height of the body of the overpack. The pedestal height is not adjusted. Conservatively, all calculations in this chapter used the shorter height for the HI-STORM 100S.

9. In February 2002, the top plate on the HI-STORM 100 overpack was modified to be two pieces in a shear ring arrangement. The total thickness of the top plate was not changed. However, there is approximately a 0.5 inch gap between the two pieces of the top plate. This gap was not modeled in MCNP since it will result in a small increase in the dose rate on the overpack lid in an area where the dose rate is greatly reduced compared to other locations on the lid.

10. The MPC base support in the HI-STORM 100S Version B was conservatively modeled as a 1 inch thick plate resting on a two inch tall ring as shown in Figure 5.3.22. The design of the overpack utilizes a solid three inch plate.

11. The gussets in the inside lower corners of the HI-STORM 100S Version B overpack were not modeled. Concrete was modeled instead.

5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

The MCNP model of the HI-TRAC transfer cask describes the lifting trunnions, pocket trunnions, and the opening in the HI-TRAC top lid. The fins through the HI-TRAC water jacket are also modeled. Streaming considerations through these trunnions and fins are discussed in Section 5.4.1.

The design of the HI-STORM 100 System, as described in the drawings in Chapter 1, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the HI-TRAC transfer cask.
- The pocket trunnions of the HI-TRAC are modeled as solid blocks of steel. No credit is taken for any part of the pocket trunnion that extends beyond the water jacket.
- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.
- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. The holes in the vent and drain port cover plates are filled with a set screw and plug weld. The steel lost in the MPC lid at the port location is replaced with a block of steel approximately 6 inches thick located directly below the port opening and attached to the underside of the lid. This design feature is shown on the drawings in Chapter 1. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid plate.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM 100 System and HI-TRAC shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

~~Since the HI-STORM 100S, 100S Version B, and the newer configuration of the HI-STORM 100 do not have the inner shield shell present, the minimum density of the concrete in the body (not the lid or pedestal) of the overpack has been increased slightly to compensate for the change in shielding relative to the HI-STORM 100 overpack with the inner shield shell. Table 5.3.2 shows the concrete composition and densities that were used for the HI-STORM 100 and HI-STORM 100S overpacks. Since the density of concrete is increased by altering the aggregate that is used, the composition of the slightly denser concrete was calculated by keeping the same mass of water as the 2.35 gm/cc composition and increasing all other components by the same ratio.~~

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

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

~~Table 4.4.36~~*Section 4.4* indicates that there are localized areas in the concrete in the lid of the overpack which approach ~~339~~*390*°F. ~~A bounding~~*An* increase in temperature from 300°F to ~~365~~*390*°F results in an approximate 0.~~424~~*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.~~555~~*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 ~~365°F~~ *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 is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
BWR					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm³)	Elements	Mass Fraction (%)
Uranium Oxide	10.412	^{235}U	2.9971(BWR) 3.2615(PWR)
		^{238}U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral†	2.644	^{10}B	4.4226 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM)4.367 (MPC-24 in HI-TRAC)
		^{11}B	20.1474 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

† All B-10 loadings in the Boral compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		^{10}B	0.14087
		^{11}B	0.64174
BWR Fuel Region Mixture	4.29251	^{235}U	2.4966
		^{238}U	70.9315
		O	9.8709
		Zr	16.4046
		N	8.35E-05
		Cr	0.0167
		Fe	0.0209
		Sn	0.2505
PWR Fuel Region Mixture	3.869939	^{235}U	2.7652
		^{238}U	71.9715
		O	10.0469
		Zr	14.9015
		Cr	0.0198
		Fe	0.0365
		Sn	0.2587

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.4862	SS304	100
Gas Plenum Springs (BWR)	0.2653	SS304	100
Expansion Springs (BWR)	0.6775	SS304	100
Upper End Fitting (BWR)	1.3692	SS304	100
Handle (BWR)	0.2572	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.9140 (water jacket)	H	11.2
	0.9619 (inside MPC)	O	88.8

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Concrete	2.~~35~~*24*	H	0.6
	~~Lid and pedestal of the~~	O	50.0
	~~HI-STORM 100, 100S, and 100S Version B~~	Si	31.5
	~~and the body of the 100~~	Al	4.8
	~~when the inner shield~~	Na	1.7
	~~shell is present~~	Ca	8.3
		Fe	1.2
		K	1.9
~~Concrete~~	~~2.48~~	~~H~~	~~0.569~~
	~~HI-STORM 100S and 100S Version B body~~	~~O~~	~~49.884~~
	~~and HI-STORM 100 body~~	~~Si~~	~~31.594~~
	~~when the inner shield shell~~	~~Al~~	~~4.814~~
	~~is not present~~	~~Na~~	~~1.705~~
		~~Ca~~	~~8.325~~
		~~Fe~~	~~1.204~~
		~~K~~	~~1.905~~
Soil	*1.7*	*H*	*0.962*
		O	*54.361*
		Al	*12.859*
		Si	*31.818*

Table 5.3.3

COMPOSITION OF THE FUEL PELLETS IN THE MIXED OXIDE FUEL ASSEMBLIES

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Mixed Oxide Pellets	10.412	^{238}U	85.498
		^{235}U	0.612
		^{238}Pu	0.421
		^{239}Pu	1.455
		^{240}Pu	0.034
		^{241}Pu	0.123
		^{242}Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	^{238}U	86.175
		^{235}U	1.975
		O	11.85

5.4 SHIELDING EVALUATION

The MCNP-4A code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.11 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.11 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.C. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

The dose rate at the various locations were calculated with MCNP using a two step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in the fuel and each axial location in the end fittings. The second and last step was to multiply the dose rate per starting particle for each group or starting location by the source strength (i.e. particles/sec) in that group or location and sum the resulting dose rates for all groups in each dose location. The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location.

The HI-STORM shielding analysis was performed for conservative burnup and cooling time combinations which bound the uniform and regionalized loading specifications for zircaloy clad fuel specified in Section 2.1.9. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

Tables ~~5.1.1 through 5.1.3 and~~ 5.1.11 through 5.1.13 provide the maximum dose rates adjacent to the HI-STORM overpack during normal conditions for each of the MPCs. Tables ~~5.1.4 through 5.1.6 and~~ 5.1.14 through 5.1.16 provide the maximum dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Sections 5.1.1 and 5.1.2.

Tables 5.1.7 and 5.1.8 provide dose rates for the 100-ton and 125-ton HI-TRAC transfer casks, respectively, with the MPC-24 loaded with design basis fuel in the normal condition, in which the MPC is dry and the HI-TRAC water jacket is filled with water. Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with the water jacket filled with water (condition in which welding operations are performed). Dose locations 4 and 5, which are on the top and bottom of the HI-TRAC were not calculated at the one-meter distance for these configurations. For the conditions involving a fully flooded MPC, the internal water level was 10 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.7 indicates that the dose rates in the upper and lower portions of the HI-TRAC are reduced by about 50% with the water in the MPC. The dose at the center of the HI-TRAC is reduced by approximately 50% when there is also water in the water jacket and is essentially unchanged when there is no water in the water jacket as compared to the normal condition results shown in Table 5.1.7.

The burnup and cooling time combination of 46*60*,000 MWD/MTU and 3 years was selected for the 100-ton MPC-24 HI-TRAC analysis because this combination of burnup and cooling time results in the highest dose rates, and therefore, bounds all other requested combinations in the 100-ton HI-TRAC. ~~For comparison, dose rates corresponding to a burnup of 75,000 MWD/MTU and 5 year cooling time for the MPC-24 are provided in Table 5.4.4.~~ The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results *in Table 5.1.7* clearly indicate that *gammas are the dominant portion of the total dose rate. Therefore,* as the burnup and cooling time increases, the reduction in the gamma dose rate due to the increased cooling time results in a net decrease in the total dose rate. ~~This result is due to the fact that the dose rates surrounding the 100-ton HI-TRAC transfer cask are gamma dominated.~~

In contrast, the dose rates surrounding the HI-TRAC 125 and 125D transfer casks have significantly higher neutron component. Therefore, the dose rates at 75,000 MWD/MTU burnup and 5 year cooling are higher than the dose rates at ~~46~~*60*,000 MWD/MTU burnup and 3 year cooling. The dose rates for the 125-ton HI-TRACs with the MPC-24 at 75,000 MWD/MTU and 5 year cooling are listed in Table 5.1.8 of Section 5.1. ~~For comparison, dose rates corresponding to a burnup of 46,000 MWD/MTU and 3 year cooling time for the MPC-24 are provided in Table 5.4.5.~~

Tables 5.4.9 ~~and 5.4.10~~ provide*s* dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-68 at *a* burnup and cooling time combinations of ~~39~~*50*,000 MWD/MTU and 3 years~~and 70,000 MWD/MTU and 6 years, respectively~~. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-68 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-68 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Tables 5.4.11 ~~and 5.4.12~~ provide*s* dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-32 at burnup and cooling time combinations of ~~35~~*45*,000 MWD/MTU and 3 years~~and 75,000 MWD/MTU and 8 years, respectively~~. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top of the 100-ton HI-TRAC are somewhat higher in the MPC-32 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-32 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

As mentioned in Section 5.0, all MPCs offer a regionalized loading pattern as described in Section 2.1.9. This loading pattern authorizes fuel of higher decay heat than uniform loading (i.e. higher burnups and shorter cooling times) to be stored in *either* the center region, region 1, of the MPC *or*~~.~~ ~~T~~*t*he outer region, region 2~~, of the MPC in regionalized loading is authorized to store fuel of lower decay heat than uniform loading (i.e. lower burnups and longer cooling times)~~. From a shielding perspective, *placing* the older fuel on the outside provides shielding for the inner fuel in the radial direction. ~~Regionalized patterns were specifically analyzed in each MPC in the 100-ton HI-TRAC.~~ Based on analysis using the same burnup and cooling times in region 1 and 2 the following percentages were calculated for dose location 2 on the 100-ton HI-TRAC.

- Approximately 21%, 27%, and 8% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively. Region 1 contains 12 (38% of total), 32 (47% of total), and 4 (17% of total) assemblies in the MPC-32, MPC-68, and MPC-24 respectively.

- Approximately 1%, 2%, and 0.2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively.

These results clearly indicate that the outer fuel assemblies shield almost all of the gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. In the axial direction, regionalized loading *with higher burnup fuel on the inside* results in higher dose rates in the center portion of the cask since the region 2 assemblies are not shielding the region 1 assemblies for axial dose locations.

~~Bounding b~~*B*urnup and cooling time combinations ~~for~~ *which bound both* regionalized loading *and uniform loading patterns* were analyzed~~and compared to the dose rates from uniform loading patterns. It was concluded that, in general, the radial dose rates from regionalized loading are bounded by the radial dose rates from uniform loading patterns~~. Therefore, dose rates for specific regionalized loading patterns are not presented in this chapter. ~~In the axial direction, the reverse may be true since the inner fuel assemblies in a regionalized loading pattern have a higher burnup than the assemblies in the uniform loading patterns. However, as depicted in the graphical data in Section 5.1.1, the dose rate along the pool or transfer lids decrease substantially moving radially outward from the center of the lid. Therefore, this increase in the dose rate in the center of the lids due to regionalized loading does not significantly impact the occupational exposure.~~ Section 5.4.9 provides *a brief* additional discussion on regionalized loading~~dose rates compared to uniform loading dose rates~~.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations

The HI-STORM 100 overpack and the HI-TRAC utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose. In addition to the fins, the pocket trunnions in the HI-TRAC 100 and 125 are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Analysis of the pocket trunnions in the HI-TRAC 100 and 125 and the steel fins in the HI-TRAC 100, 125, and 125D indicate that neutron streaming is noticeable at the surface of the transfer cask. The neutron dose rate on the surface of the pocket trunnion is approximately 5 times higher than the circumferential average dose rate at that location. The gamma dose rate is approximately 10 times lower than the circumferential average dose rate at that location. The streaming at the rib location is the largest in the HI-TRAC 125D because the ribs are thicker than in the HI-TRAC 100 or 125. The neutron dose rate on the surface of the rib in the 125D is approximately 3 times higher than the circumferential average dose rate at that location. The gamma dose rate on the surface of the rib in the 125D is approximately 3 times lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins and the pocket trunnions compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask only surface average dose rates are reported in this chapter.

Below each lifting trunnion, there is a localized area where the water jacket has been reduced in height by 4.125 inches to accommodate the lift yoke (see Figures 5.3.12 and 5.3.13). This area experiences a significantly higher than average dose rate on contact of the HI-TRAC. The peak dose in this location is 2.~~6~~*9* Rem/hr for the MPC-32, ~~1.9~~*2.0* Rem/hr for the MPC-68 and 2.4 Rem/hr for the MPC-24 in the 100-ton HI-TRAC and 1.7 Rem/hr for the MPC-24 in the HI-TRAC 125D. At a distance of 1 to 2 feet from the edge of the HI-TRAC the localized effect is greatly reduced. This dose rate is acceptable because during lifting operations the lift yoke will be in place, which, due to the additional lift yoke steel (~3 inches), will greatly reduce the dose rate. However, more importantly, people will be prohibited from being in the vicinity of the lifting trunnions during lifting operations as a standard rigging practice. In addition the lift yoke is remote in its attachment and detachment, further minimizing personnel exposure. Immediately following the detachment of the lift yoke, in preparation for closure operations, temporary shielding may be placed in this area. Any temporary shielding (e.g., lead bricks, water tanks, lead blankets, steel plates, etc.) is sufficient to attenuate the localized hot spot. The operating procedure in Chapter 8 discusses the placement of temporary shielding in this area. For the 100-ton HI-TRAC, the optional temporary shield ring will replace the water that was lost from the axial reduction in the water jacket thereby eliminating the localized hot spot. When the HI-TRAC is in the horizontal position, during transport operations, it will (at a minimum) be positioned a few feet off the ground by the transport vehicle and therefore this location below the lifting trunnions will be positioned above people which will minimize the effect on personnel exposure. In addition, good operating practice will dictate that personnel remain at least a few feet away from the transport vehicle. During vertical transport of a loaded HI-TRAC, the localized hot spot will be even further from the operating personnel. Based on these considerations, the conclusion is that this localized hot spot does not significantly impact the personnel exposure.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed that the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Chapter 1, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Dividing the total fuel gamma source for a 6x6 fuel assembly in Table 5.2.7 by the 80 inch rubble height provides a gamma source per inch of 3.41E+12 photon/s. Dividing the total neutron source for a 6x6 fuel assembly in Table 5.2.18 by 80 inches provides a neutron source per inch of 2.75E+05 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.08E+13 photon/s and 9.17E+05 neutron/s, respectively, for a burnup and cooling time of 40,000 MWD/MTU and 5 years. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling by the active fuel length of 144 inches. Therefore, damaged Dresden 1 and Humboldt Bay fuel assemblies are bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates from Dresden 1 or Humboldt Bay fuel assemblies are provided as they are bounded by the intact fuel analysis.

5.4.2.2 Generic PWR and BWR Damaged Fuel

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact. As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was performed to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Only the 100-ton HI-TRAC was analyzed because it can be concluded that if the dose rate change is not significant for the 100-ton HI-TRAC then the change will not be significant for the 125-ton HI-TRACs or the HI-STORM overpacks.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density which is higher than the fuel density for an intact fuel assembly. If the damaged fuel assembly were to fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. This scenario was analyzed with MCNP-4A in a conservative bounding fashion to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 (4 peripheral locations in the MPC-24E or MPC-24EF) and the MPC-68 (16 peripheral locations) with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

Tables 5.4.13 and 5.4.14 provide the results for the MPC-24 and MPC-68, respectively. Only the radial dose rates are provided since the axial dose rates will not be significantly affected because the damaged fuel assemblies are located on the periphery of the baskets. A comparison of these results to the results in Tables 5.1.7 and 5.4.9 indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase by less than ~~20~~*27*% while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the bottom and top is due to the assumed flat power distribution. The dose rates shown in Tables 5.4.13 and 5.4.14 were averaged over the circumference of the cask. Since almost all of the peripheral cells in the MPC-68 are filled with DFCs, an azimuthal variation would not be expected for the MPC-68. However, since there are only 4 DFCs in the MPC-24E, an azimuthal variation in dose due to the damaged fuel/fuel debris might be expected. Therefore, the dose rates were evaluated in four smaller regions, one outside each DFC, that encompass about 44% of the circumference. There was no significant change in the dose rate as a result of the localized dose calculation. These results indicate that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter.

The MPC-32 was not explicitly analyzed for damaged fuel or fuel debris in this chapter. However, based on the analysis described above for the MPC-24 and the MPC-68, it can be concluded that the shielding performance of the MPC-32 will not be significantly affected by the storage of damaged fuel.

5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single HI-STORM overpack loaded with an MPC-24 and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters were evaluated with MCNP. In the model, the casks were placed on an infinite slab of dirt to account for earth-shine effects. The atmosphere was represented by dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 700 meters. This is more than sufficient to properly account for skyshine effects. The models included either 500 or 1050 meters of air around the cask. Based on the behavior of the dose rate as a function of distance, 50 meters of air, beyond the detector locations, is sufficient to account for back-scattering. Therefore, the HI-STORM MCNP off-site dose models account for back scattering by including more than 50 meters of air beyond the detector locations for all cited dose rates. Since gamma back-scattering has an effect on the off-site dose, it is recommended that the site-specific evaluation under 10CFR72.212 include at least 50 to 100 meters of air, beyond the detector locations, in the calculational models.

The MCNP calculations of the off-site dose used a two-stage process. In the first stage a binary surface source file (MCNP terminology) containing particle track information was written for particles crossing the outer radial and top surfaces of the HI-STORM overpack. In the second stage of the calculation, this surface source file was used with the particle tracks originating on the outer edge of the overpack and the dose rate was calculated at the desired location (hundreds of meters away from the overpack). The results from this two-stage process are statistically the same as the results from a single calculation. However, the advantage of the two-stage process is that each stage can be optimized independently.

The annual dose, assuming 100% occupancy (8760 hours), at ~~300~~*350* meters from a single HI-STORM 100S Version B cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed. This table indicates that the dose due to neutrons is 2.~~5~~*8* % of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques may not properly account for the neutron transmissions and could lead to low estimates of the site boundary dose.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM 100S Version B overpack was calculated at the distance desired. Dose value = A.

2. The annual dose from the radiation leaving the top of the HI-STORM 100S Version B overpack was calculated at the distance desired. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STORM 100S Version B overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7 for the bounding burnup and cooling time of ~~47,500~~*60,000* MWD/MTU and 3-year cooling. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM 100S Version B overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

As an example, the dose from a 2x3 array at ~~400~~*450* meters is presented.

1. The annual dose from the side of a single cask: Dose A = ~~4.71~~*6.81*
2. The annual dose from the top of a single cask: Dose B = 1.~~86E~~*78E*-2
3. The annual dose from the side of a cask positioned behind another cask: Dose C = 0.~~94~~*1.36*

Using the formula shown above (Z=3), the total dose at ~~400~~*450* meters from a 2x3 array of HI-STORM overpacks is ~~17.06~~*24.62* mrem/year, assuming a 8760 hour occupancy.

An important point to notice here is that the dose from the side of the back row of casks is approximately 16 % of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis. Therefore, a site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

5.4.4 Stainless Steel Clad Fuel Evaluation

Table 5.4.8 presents the dose rates at the center of the HI-STORM 100 overpack, adjacent and at one meter distance, from the stainless steel clad fuel. These dose rates, when compared to Tables 5.1.1 through 5.1.6, are similar to the dose rates from the design basis zircaloy clad fuel, indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STORM 100 overpack. Since the true active fuel length is shorter than 144 inches and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

5.4.5 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the source terms for the BWR design basis zircaloy clad fuel assembly (GE 7x7) which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.22 by the 110 inch active fuel height provides a gamma source per inch of 2.36E+12 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.23 by 110 inches provides a neutron source strength per inch of 3.06E+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.08E+13 photons/s and 9.17E+5 neutrons/s for 40,000 MWD/MTU and 5 year cooling. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling by the active fuel length of 144 inches. This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Section 5.4.2.1, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches. The resulting gamma and neutron source strengths are 3.25E+12 photons/s and 4.21E+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

5.4.6 Non-Fuel Hardware

As discussed in Section 5.2.4, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM 100 System. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. BPRAs,~~-and~~ TPDs, *and CRAs* are authorized for unrestricted storage in an MPC while the ~~CRAs and~~ APSRs are restricted to the center four locations in the MPC-24, MPC-24E, MPC-24EF and *the center twelve locations in the* MPC-32. The calculation of the source term and a description of the bounding fuel devices was provided in Section 5.2.4. The dose rate due to BPRAs and TPDs being stored in a fuel assembly was explicitly calculated. Table 5.4.15 provides the dose rates at various locations on the surface and one meter from the 100-ton HI-TRAC due to the BPRAs and TPDs for the MPC-24 and MPC-32. These results were added to the totals in the other table to provide the total dose rate with BPRAs. Table 5.4.15 indicates that the dose rates from BPRAs bound the dose rates from TPDs.

As discussed in Section 5.2.4, two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs. The dose rate due to CRAs and APSRs ~~being stored in the inner four fuel locations~~ was explicitly calculated for dose locations around the 100-ton HI-TRAC. Tables 5.4.16 and 5.4.17 provide the results for the different configurations of CRAs and APSRs, respectively, in the MPC-24 and MPC-32. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is ~~minimal~~ *less than the dose rate from BPRAs* and the dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, *these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.* ~~as noted in Tables 5.4.16 and 5.4.17, the dose rate at the edge of the transfer lid is almost negligible due to APSRs and CRAs.~~ Therefore, even though the dose rates calculated (using a very conservative source term evaluation) are ~~daunting~~ *very high,* they do not pose a risk from an operations perspective because they are localized in nature. Section 5.1.1 provides additional discussion on the acceptability of the relatively high localized doses on the bottom of the HI-TRACs.

5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately 1E+8 neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as 5.8E-6 neutrons per gamma (1E+8/865/3.7E+10/0.54) with energy greater than 1.666 MeV or 1.16E+5 neutrons/curie (1E+8/865) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

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

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

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

5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.37 and 5.2.7 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.18, bounds the thoria rod neutron spectra, Table 5.2.38, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the 100-ton HI-TRAC and the HI-STORM overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher for the 100-ton HI-TRAC and only 15% lower for the HI-STORM overpack than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

5.4.9 Regionalized Loading Dose Rate Evaluation

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.

~~Regionalized loading patterns for the MPC-24, MPC-32, and MPC-68 are considered in this section.. Burnup and cooling time combinations bounding the 14x14A and 9x9G array classes were used in the analysis since for uniform loading these array classes have the highest permissible burnup for a given cooling time. Section 2.1.9 describes the calculation of the allowable burnup and cooling times for regionalized loading. Rather than explicitly analyzing regionalized loading patterns, uniform loading burnup and cooling time combinations which bound the regionalized values were analyzed in this section. The dose rates from these bounding uniform patters were compared to the uniform dose rates reported in this chapter.~~

~~It was determined that for the MPC-32, all radial 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated dose rates in the axial locations for regionalized loading were less than 10% higher than the uniform dose rates reported in this chapter at 1 meter from the overpack.~~

~~For the MPC-24 and MPC-68 it was determined that all 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter.~~

~~Based on these results it can be stated that regionalized loading patterns will reduce the dose rate in the radial direction by shielding the hotter fuel on the inside of the cask with colder fuel on the outside of the cask. However, in the axial direction the localized dose rates in the center of the cask may increase as a result of the regionalized loading pattern. This is a localized effect, which has dissipated at the edge of the cask, and therefore will not result in a significant increase to the occupational exposure rates. In addition, it should be mentioned that the localized increase on the bottom center of the overpack is an area where workers will normally not be present and the increase in the top center of the overpack is an area where workers minimize their stay.~~

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Neutron Energy (MeV)	Quality Factor	(rem/hr)[†]/(n/cm^2-s)
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
4660,000 MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	44.75	370.31	55.57	470.63	473.46
2	2649.74	0.79	810.25	3460.78	3664.65
3	9.98	484.21	11.35	505.54	637.13
4	39.13	367.92	2.10	409.15	511.41
5 (pool lid)	126.99	2071.23	5.62	2203.84†††	2214.03
ONE METER FROM THE 100-TON HI-TRAC					
1	351.97	74.73	115.53	542.22	568.65
2	1171.07	7.08	265.57	1443.73	1533.97
3	140.14	117.78	47.64	305.55	349.87

Note: MPC internal water level is 10 inches below the MPC lid.

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.3

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
4660,000 MWD/MTU AND 3-YEAR COOLING

Dose Point[†] Location	**Fuel Gammas[††] (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC					
1	*34.56*	*318.37*	*6.08*	*359.01*	*361.11*
2	*1565.21*	*0.50*	*53.06*	*1618.77*	*1734.33*
3	*6.38*	*483.03*	*1.07*	*490.48*	*621.41*
4	*39.09*	*367.91*	*1.93*	*408.92*	*511.18*
5 (pool lid)	*126.63*	*2071.47*	*5.41*	*2203.51*[†††]	*2213.68*
ONE METER FROM THE 100-TON HI-TRAC					
1	*202.31*	*49.47*	*7.09*	*258.86*	*273.65*
2	*681.82*	*3.93*	*19.97*	*705.72*	*756.28*
3	*80.54*	*81.26*	*2.41*	*164.21*	*193.09*

Note: MPC internal water level is 10 inches below the MPC lid.

[†] Refer to Figures 5.1.2 and 5.1.4.

[††] Gammas generated by neutron capture are included with fuel gammas.

[†††] Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.4

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~~DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT~~
~~75,000 MWD/MTU AND 5-YEAR COOLING~~

~~Dose Point Location~~	~~Fuel Gammas (mrem/hr)~~	~~(n,γ) Gammas (mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~ADJACENT TO THE 100-TON HI-TRAC~~						
~~1~~	~~61.51~~	~~59.98~~	~~841.88~~	~~848.87~~	~~1812.25~~	~~1820.79~~
~~2~~	~~1720.78~~	~~244.11~~	~~0.84~~	~~450.27~~	~~2416.00~~	~~2663.24~~
~~3~~	~~16.84~~	~~11.76~~	~~464.19~~	~~710.32~~	~~1203.11~~	~~1351.64~~
~~3 (temp)~~	~~7.62~~	~~20.93~~	~~215.15~~	~~11.42~~	~~255.11~~	~~323.26~~
4	~~41.62~~	~~4.64~~	~~373.59~~	~~874.50~~	~~1294.35~~	~~1418.85~~
~~4 (outer)~~	~~11.60~~	~~2.95~~	~~93.02~~	~~590.24~~	~~697.81~~	~~729.14~~
~~5 (pool lid)~~	~~298.84~~	~~85.64~~	~~4241.72~~	~~5701.99~~	~~10328.19~~	~~10392.96~~
~~5 (transfer)~~	~~732.62~~	~~4.69~~	~~6320.81~~	~~3264.99~~	~~10323.11~~	~~10419.95~~
~~5(t-outer)~~	~~178.17~~	~~1.60~~	~~611.80~~	~~1290.11~~	~~2081.69~~	~~2103.16~~
~~ONE METER FROM THE 100-TON HI-TRAC~~						
~~1~~	~~226.79~~	~~32.24~~	~~125.15~~	~~137.98~~	~~522.16~~	~~554.64~~
~~2~~	~~754.47~~	~~74.62~~	~~9.90~~	~~168.82~~	~~1007.81~~	~~1117.29~~
~~3~~	~~94.60~~	~~17.96~~	~~103.96~~	~~66.26~~	~~282.78~~	~~332.22~~
~~3 (temp)~~	~~94.09~~	~~19.29~~	~~88.55~~	~~25.04~~	~~226.97~~	~~271.54~~
4	~~14.19~~	~~0.81~~	~~115.34~~	~~217.83~~	~~348.17~~	~~386.74~~
~~5 (transfer)~~	~~315.47~~	~~0.86~~	~~2582.07~~	~~911.26~~	~~3809.66~~	~~3848.79~~
~~5(t-outer)~~	~~42.95~~	~~2.78~~	~~232.74~~	~~261.61~~	~~540.08~~	~~543.98~~

~~Notes:~~

- ~~Refer to Figures 5.1.2 and 5.1.4 for dose locations.~~
- ~~Dose location 3(temp) represents dose location 3 with temporary shielding installed.~~
- ~~Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.~~
- ~~Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.~~
- ~~Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.~~

Table 5.4.5

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~~DOSE RATES FROM THE 125-TON HI-TRAC FOR NORMAL CONDITIONS MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT 46,000 MWD/MTU AND 3-YEAR COOLING~~

~~Dose Point Location~~	~~Fuel Gammas (mrem/hr)~~	~~(n,γ) Gammas (mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~ADJACENT TO THE 125-TON HI-TRAC~~						
~~1~~	~~12.43~~	~~17.84~~	~~101.50~~	~~119.96~~	~~251.73~~	~~252.44~~
~~2~~	~~211.88~~	~~52.83~~	~~0.01~~	~~83.09~~	~~347.81~~	~~363.68~~
~~3~~	~~2.66~~	~~1.89~~	~~62.80~~	~~191.39~~	~~258.73~~	~~278.44~~
~~4~~	~~71.16~~	~~2.42~~	~~343.61~~	~~221.50~~	~~638.70~~	~~754.12~~
~~4 (outer)~~	~~9.28~~	~~1.73~~	~~42.67~~	~~4.65~~	~~58.33~~	~~72.52~~
~~5 (pool)~~	~~108.22~~	~~0.90~~	~~529.32~~	~~766.89~~	~~1405.34~~	~~1413.04~~
~~5 (transfer)~~	~~112.43~~	~~1.38~~	~~606.59~~	~~127.01~~	~~847.40~~	~~852.89~~
~~ONE METER FROM THE 125-TON HI-TRAC~~						
~~1~~	~~28.53~~	~~7.12~~	~~13.01~~	~~19.74~~	~~68.40~~	~~70.43~~
~~2~~	~~95.52~~	~~17.13~~	~~0.53~~	~~28.34~~	~~141.51~~	~~148.58~~
~~3~~	~~10.94~~	~~4.02~~	~~12.69~~	~~17.61~~	~~45.26~~	~~50.18~~
~~4~~	~~20.01~~	~~0.58~~	~~82.73~~	~~22.81~~	~~126.13~~	~~153.78~~
~~5 (transfer)~~	~~41.40~~	~~0.27~~	~~293.26~~	~~22.00~~	~~356.92~~	~~359.85~~

~~Notes:~~

- ~~Refer to Figures 5.1.2 and 5.1.4 for dose locations.~~
- ~~Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.~~
- ~~Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.~~

Table 5.4.6

ANNUAL DOSE AT ~~300~~ *350* METERS FROM A SINGLE HI-STORM 100S VERSION B OVERPACK WITH AN MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL†

Dose Component	~~47,500~~ *60,000* MWD/MTU 3-Year Cooling (mrem/yr)
Fuel gammas††	*17.54*
^{60}Co Gammas	*1.18*
Neutrons	*0.54*
Total	*19.26*

† 8760 hour annual occupancy is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS HI-STORM 100S VERSION B ISFSI CONFIGURATIONS ~~47,500~~*60,000* MWD/MTU AND 3-YEAR COOLING ZIRCALOY CLAD FUEL†

Distance	A Side of Overpack (mrem/yr)	B Top of Overpack (mrem/yr)	C Side of Shielded Overpack (mrem/yr)
100 meters	*880.47*	*2.79*	*176.09*
150 meters	*299.26*	*1.04*	*59.85*
200 meters	*129.22*	*0.48*	*25.84*
250 meters	*64.50*	*0.22*	*12.90*
300 meters	*34.11*	*0.12*	*6.82*
350 meters	*19.20*	*6.06E-02*	*3.84*
400 meters	*11.26*	*3.25E-02*	*2.25*
450 meters	*6.81*	*1.78E-02*	*1.36*
500 meters	*4.22*	*1.07E-02*	*0.84*
550 meters	*2.71*	*6.94E-03*	*0.54*
600 meters	*1.74*	*4.13E-03*	*0.35*

† 8760 hour annual occupancy is assumed.

Table 5.4.8

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR DESIGN BASIS STAINLESS STEEL CLAD FUEL WITHOUT BPRAs

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
MPC-24 (40,000 MWD/MTU AND 8-YEAR COOLING)				
2 (Adjacent)	36.97	0.02	1.11	38.10
2 (One Meter)	18.76	0.17	0.50	19.43
MPC-32 (40,000 MWD/MTU AND 9-YEAR COOLING)				
2 (Adjacent)	37.58	0.00	1.49	39.08
2 (One Meter)	18.74	0.25	0.58	19.57
MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)				
2 (Adjacent)	17.79	0.01	0.10	17.90
2 (One Meter)	8.98	0.13	0.04	9.15

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.9

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~39~~*50*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*118.31*	*25.51*	*1260.91*	*340.66*	*1745.39*
2	*2638.99*	*125.79*	*0.80*	*227.40*	*2992.98*
3	*7.38*	*2.57*	*833.34*	*147.88*	*991.17*
3 (temp)	*4.33*	*4.53*	*408.34*	*2.70*	*419.89*
4	*25.26*	*0.90*	*494.35*	*195.30*	*715.82*
4 (outer)	*7.49*	*0.70*	*132.37*	*124.46*	*265.02*
5 (pool lid)	*351.88*	*31.12*	*5646.34*	*2040.15*	*8069.50*
5(transfer lid)	*509.02*	*1.43*	*8507.35*	*1288.83*	*10306.63*
5 (t-outer)	*187.09*	*0.61*	*750.08*	*479.11*	*1416.90*
ONE METER FROM THE 100-TON HI-TRAC					
1	*359.24*	*15.46*	*117.76*	*60.21*	*552.67*
2	*1134.59*	*35.59*	*8.69*	*78.41*	*1257.27*
3	*88.49*	*6.19*	*178.43*	*16.85*	*289.97*
3 (temp)	*88.38*	*6.52*	*142.94*	*8.14*	*245.97*
4	*7.94*	*0.52*	*151.01*	*47.61*	*207.07*
5(transfer lid)	*260.67*	*0.62*	*3774.45*	*344.08*	*4379.82*
5 (t-outer)	*32.63*	*1.12*	*318.78*	*97.39*	*449.92*

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.10

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~~DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS~~
~~MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT~~
~~70,000 MWD/MTU AND 6-YEAR COOLING~~

~~Dose Point Location~~	~~Fuel Gammas (mrem/hr)~~	~~(n,γ) Gammas (mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~
~~ADJACENT TO THE 100-TON HI-TRAC~~					
~~1~~	~~46.14~~	~~51.57~~	~~957.08~~	~~688.55~~	~~1743.33~~
~~2~~	~~1122.52~~	~~254.29~~	~~0.61~~	~~459.52~~	~~1836.95~~
~~3~~	~~2.41~~	~~5.19~~	~~632.54~~	~~298.91~~	~~939.05~~
~~3 (temp)~~	~~1.55~~	~~9.15~~	~~309.94~~	~~5.45~~	~~326.10~~
4	~~9.75~~	~~1.83~~	~~375.23~~	~~394.76~~	~~781.56~~
~~4 (outer)~~	~~2.71~~	~~1.42~~	~~100.47~~	~~251.59~~	~~356.19~~
~~5 (pool lid)~~	~~138.45~~	~~62.92~~	~~4285.78~~	~~4123.84~~	~~8610.99~~
~~5(transfer lid)~~	~~239.62~~	~~2.89~~	~~6457.39~~	~~2605.31~~	~~9305.22~~
~~5 (t-outer)~~	~~81.19~~	~~1.24~~	~~569.34~~	~~968.46~~	~~1620.24~~
~~ONE METER FROM THE 100-TON HI-TRAC~~					
~~1~~	~~151.92~~	~~31.25~~	~~89.38~~	~~121.68~~	~~394.23~~
~~2~~	~~480.13~~	~~71.95~~	~~6.59~~	~~158.43~~	~~717.11~~
~~3~~	~~37.30~~	~~12.51~~	~~135.43~~	~~34.06~~	~~219.31~~
~~3 (temp)~~	~~37.24~~	~~13.17~~	~~108.50~~	~~16.44~~	~~175.35~~
4	~~2.87~~	~~1.05~~	~~114.62~~	~~96.22~~	~~214.76~~
~~5(transfer lid)~~	~~116.13~~	~~1.26~~	~~2864.94~~	~~695.54~~	~~3677.88~~
~~5 (t-outer)~~	~~14.24~~	~~2.26~~	~~241.96~~	~~196.86~~	~~455.33~~

~~Notes:~~
- ~~Refer to Figures 5.1.2 and 5.1.4 for dose locations.~~
- ~~Dose location 3(temp) represents dose location 3 with temporary shielding installed.~~
- ~~Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.~~
- ~~Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.~~
- ~~Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.~~

Table 5.4.11

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
~~35~~45,000 MWD/MTU AND 3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*123.09*	*17.99*	*1134.82*	*253.68*	*1529.57*	*1539.91*
2	*3059.28*	*77.64*	*1.63*	*145.58*	*3284.12*	*3576.87*
3	*42.51*	*3.43*	*703.72*	*203.47*	*953.12*	*1178.63*
4	*102.68*	*1.55*	*527.34*	*253.44*	*885.02*	*1064.21*
4 (outer)	*28.17*	*0.91*	*132.30*	*174.27*	*335.65*	*380.68*
5 (pool)	*745.11*	*26.66*	*6237.41*	*1665.29*	*8674.47*	*8757.27*
5 (transfer)	*1366.50*	*1.04*	*9401.97*	*966.70*	*11736.20*	*11837.01*
5(t-outer)	*246.52*	*0.48*	*787.00*	*363.83*	*1397.82*	*1418.02*
ONE METER FROM THE 100-TON HI-TRAC						
1	*407.43*	*10.23*	*168.91*	*42.45*	*629.02*	*667.20*
2	*1350.21*	*24.30*	*12.79*	*54.21*	*1441.51*	*1571.36*
3	*176.77*	*5.64*	*144.81*	*19.97*	*347.18*	*414.23*
4	*31.95*	*0.40*	*157.25*	*63.15*	*252.75*	*305.95*
5 (transfer)	*567.40*	*0.23*	*3723.34*	*260.88*	*4551.86*	*4598.19*
5(t-outer)	*58.97*	*0.96*	*330.25*	*76.48*	*466.67*	*471.29*

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.12

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~~DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS~~
~~MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL~~
~~75,000 MWD/MTU AND 8-YEAR COOLING~~

~~Dose Point Location~~	~~Fuel Gammas (mrem/hr)~~	~~(n,γ) Gammas (mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~
~~ADJACENT TO THE 100-TON HI-TRAC~~						
~~1~~	~~35.68~~	~~60.59~~	~~770.69~~	~~854.32~~	~~1721.28~~	~~1731.62~~
~~2~~	~~1013.34~~	~~261.53~~	~~1.10~~	~~490.12~~	~~1766.09~~	~~2058.84~~
~~3~~	~~10.86~~	~~11.54~~	~~477.92~~	~~685.26~~	~~1185.58~~	~~1411.08~~
~~4~~	~~29.98~~	~~5.24~~	~~358.13~~	~~853.47~~	~~1246.82~~	~~1426.01~~
~~4 (outer)~~	~~7.53~~	~~3.06~~	~~89.85~~	~~586.93~~	~~687.38~~	~~732.40~~
~~5 (pool)~~	~~249.21~~	~~89.80~~	~~4236.02~~	~~5608.42~~	~~10183.44~~	~~10266.23~~
~~5 (transfer)~~	~~479.57~~	~~3.49~~	~~6385.16~~	~~3256.11~~	~~10124.33~~	~~10225.14~~
~~5(t-outer)~~	~~83.17~~	~~1.62~~	~~534.47~~	~~1225.30~~	~~1844.57~~	~~1864.76~~
~~ONE METER FROM THE 100-TON HI-TRAC~~						
~~1~~	~~131.43~~	~~34.46~~	~~114.71~~	~~142.94~~	~~423.55~~	~~461.72~~
~~2~~	~~442.51~~	~~81.87~~	~~8.69~~	~~182.51~~	~~715.58~~	~~845.43~~
~~3~~	~~56.63~~	~~18.98~~	~~98.35~~	~~67.24~~	~~241.20~~	~~308.25~~
~~4~~	~~8.55~~	~~1.35~~	~~106.79~~	~~212.67~~	~~329.36~~	~~382.56~~
~~5 (transfer)~~	~~197.49~~	~~0.79~~	~~2528.63~~	~~878.62~~	~~3605.53~~	~~3651.86~~
~~5(t-outer)~~	~~19.53~~	~~3.22~~	~~224.29~~	~~257.58~~	~~504.63~~	~~509.25~~

~~Notes:~~

- ~~Refer to Figures 5.1.2 and 5.1.4 for dose locations.~~
- ~~Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.~~
- ~~Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.~~
- ~~Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.~~

Table 5.4.13

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS
WITH FOUR DAMAGED FUEL CONTAINERS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
~~46~~*60*,000 MWD/MTU AND 3-YEAR COOLING
WITHOUT BPRAs

Dose Point† Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*162.53*	*40.07*	*958.00*	*607.90*	*1768.50*
2	*3154.37*	*145.11*	*1.14*	*264.13*	*3564.75*
3	*47.06*	*9.00*	*528.22*	*583.57*	*1167.85*
ONE METER FROM THE 100-TON HI-TRAC					
1	*449.03*	*20.58*	*142.41*	*92.37*	*704.39*
2	*1404.60*	*44.91*	*11.27*	*99.48*	*1560.26*
3	*202.22*	*11.75*	*118.30*	*54.30*	*386.56*

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.14

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS
WITH SIXTEEN DAMAGED FUEL CONTAINERS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL
~~39~~*50*,000 MWD/MTU AND 3-YEAR COOLING

Dose Point† Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC					
1	*276.49*	*36.57*	*1260.91*	*629.42*	*2203.39*
2	*2488.31*	*126.39*	*0.80*	*213.81*	*2829.31*
3	*9.10*	*5.48*	*833.34*	*324.77*	*1172.69*
ONE METER FROM THE 100-TON HI-TRAC					
1	*417.32*	*19.09*	*117.76*	*90.45*	*644.62*
2	*1091.56*	*38.82*	*8.69*	*85.73*	*1224.79*
3	*121.49*	*8.49*	*178.43*	*33.62*	*342.02*

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.15

DOSE RATES DUE TO BPRAs AND TPDs FROM THE 100-TON HI-TRAC
FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC				
1	8.54	0.00	10.34	0.01
2	247.24	0.03	292.75	0.04
3	148.53	125.75	225.50	188.14
3 (temp)	68.15	56.21	93.63	77.00
4	124.50	106.71	179.19	156.14
4 (outer)	31.33	27.12	45.02	39.33
5 (pool lid)	64.77	0.00	82.79	0.01
5(transfer lid)	96.84	0.00	100.81	0.00
5(t-outer)	21.47	0.00	20.20	0.00
ONE METER FROM THE 100-TON HI-TRAC				
1	32.48	0.18	38.18	0.24
2	109.47	1.20	129.85	1.63
3	49.43	38.93	67.05	55.11
3 (temp)	44.57	35.01	59.32	48.95
4	38.57	33.37	53.20	47.19
5(transfer lid)	39.13	0.00	46.33	0.00
5(t-outer)	3.90	0.00	4.63	0.00

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.16

DOSE RATES DUE TO CRAs FROM THE 100-TON HI-TRAC
FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC				
1	*164.33*	*29.63*	*198.02*	*36.56*
2	*15.35*	*0.42*	*15.27*	*0.41*
3	*0.03*	*0.00*	*0.04*	*0.00*
4	*0.01*	*0.00*	*0.00*	*0.00*
5 (pool lid)	*1189.62*	*223.71*	*1740.02*	*329.86*
5(transfer lid)	*1892.59*	*355.40*	*2974.99*	*558.68*
5(t-outer)	*357.66*	*57.54*	*404.90*	*71.63*
ONE METER FROM THE 100-TON HI-TRAC				
1	*238.30*	*28.03*	*285.70*	*34.39*
2	*125.09*	*10.49*	*148.51*	*12.52*
3	*1.72*	*0.18*	*2.06*	*0.21*
4	*0.01*	*0.00*	*0.00*	*0.00*
5(transfer lid)	*819.78*	*152.57*	*1156.31*	*224.75*
5(t-outer)	*86.06*	*15.23*	*113.39*	*21.17*

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.17

DOSE RATES DUE TO APSRs FROM THE 100-TON HI-TRAC
FOR NORMAL CONDITIONS

	MPC-24			MPC-32		
Dose Point Location	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 3 (mrem/hr)**	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 3 (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC						
1	12.42	2.35	12.25	*57.80*	*11.73*	*56.80*
2	0.21	0.01	9.12	*0.29*	*0.02*	*17.70*
3	0.00	0.00	0.00	*0.00*	*0.00*	*0.00*
4	0.00	0.00	0.00	*0.00*	*0.00*	*0.09*
5 (pool lid)	1996.57	371.98	1941.51	*3243.57*	*645.11*	*3698.77*
5(transfer)	3021.08	572.85	2994.54	*5522.01*	*1035.17*	*5664.10*
5(t-outer)	3.41	0.54	3.57	*33.25*	*5.97*	*31.46*
ONE METER FROM THE 100-TON HI-TRAC						
1	2.73	0.46	3.49	*11.69*	*2.22*	*11.36*
2	0.61	0.07	3.31	*1.55*	*0.20*	*6.53*
3	0.02	0.00	0.04	*0.05*	*0.01*	*0.08*
4	0.00	0.00	0.00	*0.00*	*0.00*	*0.09*
5(transfer)	458.06	84.81	444.44	*1288.04*	*240.05*	*1211.44*
5(t-outer)	17.11	3.19	17.36	*64.18*	*12.16*	*62.83*

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

Table 5.4.18

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

Assembly	**Active fuel length (inch)**	**Neutrons per sec per inch**	**Neutrons per sec per inch with Sb-Be source**	**Reference for neutrons per sec per inch**
7x7 design basis	144	9.17E+5	N/A	40 GWD/MTU and 5 year cooling
6x6 design basis	110	2.0E+5	2.6E+5	Table 5.2.18
6x6 design basis MOX	110	3.06E+5	3.66E+5	Table 5.2.23

APPENDIX 5.E

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SUPPLEMENT 5.I

SHIELDING EVALUATION OF THE HI-STORM 100U SYSTEM

5.I.0 INTRODUCTION

This supplement is focused on providing a shielding evaluation of the HI-STORM 100U system pursuant to the guidelines in NUREG-1536. The evaluation presented herein supplements those evaluations of the HI-STORM overpacks contained in the main body of Chapter 5 of this FSAR, and information in the main body of Chapter 5 that remains applicable to the HI-STORM 100U is not repeated in this supplement. To aid the reader, the sections in this supplement are numbered in the same fashion as the corresponding sections in the main body of this chapter, i.e., Sections 5.I.1 through 5.I.5 correspond to Sections 5.1 through 5.5. Tables and figures in this supplement are labeled sequentially.

5.I.1 DISCUSSION AND RESULTS

The HI-STORM 100U system differs from the HI-STORM system evaluated in the main body of this chapter only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs and HI-TRAC transfer casks are identical between the systems. All calculations, results and conclusions regarding the HI-TRAC transfer cask presented in the main body of Chapter 5 are therefore directly applicable to the HI-STORM 100U system, and no further calculations for the HI-TRAC transfer cask are presented in this supplement.

The shielding design of the HI-STORM 100U VVM is similar to the overpack designs evaluated in the main body of this chapter, with gamma shielding provided by the concrete and the steel of the module, and neutron shielding provided by the module concrete. However, the VVM is mostly located below the surface of the surrounding soil. This results in additional shielding, and a significant reduction in the directly accessible surface for the VVM compared to the other overpacks. Dose rates from a HI-STORM 100U VVM at the site boundary are therefore significantly lower than, and bounded by, dose rates from the above ground HI-STORM systems evaluated in the main body of this chapter.

Shielding analyses were performed for the HI-STORM 100U with an MPC-32 loaded with intact design basis zircaloy clad fuel assemblies. The burnup and cooling time combinations analyzed are the same combinations reported in Section 5.1 for the MPC-32. Table 5.I.1 presents the results for the burnup and cooling time combination that resulted in the maximum dose rates. 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 the dose points were averaged over the vent opening while the dose values reported at 1 meter from the dose locations were taken at the midplane of the vent. Calculations were only performed for normal conditions, since Subsections 5.1.1 and 5.1.2 concluded that off-normal and accident conditions for the HI-STORM overpack are identical or equivalent to normal conditions for the purpose of the shielding evaluation.

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 Version B with an MPC-32. Furthermore, the areas associated with the maximum dose rate for the HI-STORM 100U module are the areas of the inlets and outlets. These areas are much smaller than the area associated with the maximum dose rates for the overpack analyzed in the main part of this chapter, which is the outer radial surface of the overpack. It can therefore be concluded that the HI-STORM 100U is bounded by the HI-STORM 100 systems analyzed in the main body of this chapter, since the HI-STORM 100U has a smaller directly accessible surface, lower maximum dose rates and smaller areas associated with these maximum dose rates. Nevertheless, calculations were performed to determine the dose rate from the HI-STORM 100U at a distance of 100 meters from the inlet vent. These results, which are presented in Table 5.I.2, indicate that the HI-STORM 100U easily meets the requirements of 10CFR72.104 at 100 meters. Comparing these results to the results in Table 5.4.7 demonstrates that the off-site dose from the HI-STORM 100U is a very small fraction of the off-site dose from an above ground overpack.

Finally, as observed in Section 1.I.2, the HI-STORM 100U VVM is deliberately engineered to permit the MPC cavity to be deepened such that the MPC is located deeper inside the module. The elevation of the MPC shown in the drawings in Section 1.5 and analyzed in this supplement is the highest permitted elevation. Of course, lowering the MPC in the VVM would further reduce the radiation dose rates below those computed herein.

5.I.2 SOURCE SPECIFICATION

The analyses in this supplement are performed for intact design basis zircaloy clad fuel assemblies as described in Section 5.2.

5.I.3 MODEL SPECIFICATIONS

The shielding analyses of the HI-STORM 100U module are performed with MCNP-4A, which is the same code used for the analyses presented in the main body of this chapter.

Section 1.5 provides the drawings that describe the HI-STORM 100U System. 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. Figure 5.I.2 shows cross sectional views of the HI-STORM 100U module as it was modeled in MCNP. Note that the inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the dose calculations. Note again that the MPC is assumed to be positioned at its highest permissible elevation in relation to the inlet ducts (i.e., in the configuration shown in the drawings in Section 1.5) to maximize the calculated dose rates.

Since the HI-STORM 100U model uses principally the same MPC model as the calculations in the main body of this chapter, all figures, conservative modeling approximations, and modeling differences for the MPC shown in Section 5.3 are applicable to the calculations in this supplement. The differences between models and drawings for the module are listed and discussed here.

1. Minor penetrations in the body of the module (e.g. lift locations) are not modeled as these are small localized effects which will not affect the off-site dose rates.

2. The MPC supports and guides were conservatively neglected.

3. The closure lid cover plate was modeled as flat. This conservatively reduces the amount of concrete in the lid near the outlet vent.

4. [redacted]

Composition and densities of the various materials in Table 5.3.2 were used in the analyses.

5.I.4 SHIELDING EVALUATION

Table 5.I.1 provides dose rates adjacent to and at 1 meter distance from the HI-STORM 100U module during normal conditions for the MPC-32. These results demonstrate that the dose rates around the HI-STORM 100U are exceptionally low for the very conservative burnup and cooling time combination analyzed. These results also show that the higher dose rate at the inlet vent is reduced by more than an order of magnitude at a distance of 1 meter from the vent.

Table 5.I.2 provides the annual dose at 100 meters from a HI-STORM 100U module for the MPC-32 including the contribution from BPRAs. These results clearly demonstrate that the off-site contribution from a HI-STORM 100U is a small fraction of the off-site dose from the above ground HI-STORM overpacks.

5.I.5 REGULATORY COMPLIANCE

In summary it can be concluded that dose rates from the HI-STORM 100U module are bounded by the dose rates for the overpacks analyzed in the main body of the report. The shielding system of the HI-STORM 100U System is therefore in compliance with 10CFR72 and satisfies the applicable design and acceptance criteria including 10CFR20. Thus, the shielding evaluation presented in this supplement provides reasonable assurance that the HI-STORM 100U System will allow safe storage of spent fuel.

Table 5.I.1

DOSE RATES ADJACENT TO AND 1 METER FROM THE
HI-STORM 100U MODULE
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
69,000 MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
Surface					
1	41.70	8.24	15.08	65.02	69.43
2	3.03	1.07	4.84	8.94	9.39
One Meter					
1	3.44	0.70	1.41	5.56	5.92
3	0.93	0.44	0.92	2.30	2.49

† Refer to Figure 5.I.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.I.2

ANNUAL DOSE AT 100 METERS FROM A SINGLE HI-STORM 100U OVERPACK WITH AN MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL†

Dose Component	69,000 MWD/MTU 5-Year Cooling (mrem/yr)
Fuel gammas††	4.01
^{60}Co Gammas	1.22
Neutrons	3.27
Total	8.50

† 8760 hour annual occupancy is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

FIGURE 5.I.1; HI-STORM 100U MODULE CROSS SECTIONAL ELEVATION VIEW WITH DOSE POINT LOCATION

FIGURE 5.I.2; HI-STORM 100U MODULE CROSS SECTIONAL ELEVATION VIEW. VALUES IN BRACKETS ARE IN MILLIMETERS.

CHAPTER 6†: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STORM 100 System is consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both.

4. Criticality safety of the cask system should not rely on use of the following credits:

    a. burnup of the fuel
    b. fuel-related burnable neutron absorbers
    c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test††.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STORM 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. Analyses for the HI-STAR 100 System, which are applicable to the HI-STORM 100 System, have been previously submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

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† 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).

†† For greater credit allowance, fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STORM 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM 100 System depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24~~,~~ *and* MPC-24E~~ and MPC-24EF~~);

2. The incorporation of permanent fixed neutron-absorbing panels in the fuel basket structure;

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and

4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24~~,~~ *and* MPC-24E~~ and MPC-24EF~~, and -for loading/unloading fuel in the MPC-32~~ and MPC-32F~~.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the Boral fixed neutron absorber
- more than 90 percent of the B-10 content for the Metamic fixed neutron absorber, with comprehensive fabrication tests as described in Section 9.1.5.3.2.

The following four interchangeable basket designs are available for use in the HI-STORM 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,

- a 24-cell basket (MPC-24E) for intact and damaged PWR fuel assemblies. This is a variation of the MPC-24, with an optimized cell arrangement, increased ^{10}B content in the fixed neutron absorber and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC) *containing*~~.~~ ~~Additionally, a variation in the MPC-24E, designated MPC-24EF, is designed for intact and~~ damaged PWR fuel assemblies ~~and~~ *or* PWR fuel debris. The MPC-24E ~~and MPC-24EF are~~*is* designed for fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,

- a 32-cell basket (MPC-32), designed for intact, ~~and~~ damaged PWR fuel assemblies *and PWR fuel debris* of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading. ~~Additionally, a variation in the MPC-32, designated MPC-32F, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris.~~ And

- a 68-cell basket (MPC-68), designed for ~~both~~ intact and damaged BWR fuel assemblies *and BWR fuel debris* with a specified maximum planar-average enrichment. Additionally, *a* variations in the MPC-68, designated MPC-68F ~~and MPC-68FF~~, ~~are~~ *is* designed for *specific* intact and damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.

Two interchangeable neutron absorber materials are used in these baskets, Boral and Metamic. For Boral, 75 percent of the minimum B-10 content is credited in the criticality analysis, while for Metamic, 90 percent of the minimum B-10 content is credited, based on the neutron absorber tests specified in Section 9.1.5.3. However, the B-10 content in Metamic is chosen to be lower than the B-10 content in Boral, and is chosen so that the absolute B-10 content credited in the criticality analysis is the same for the two materials. This makes the two materials identical from a criticality perspective. This is confirmed by comparing results for a selected number of cases that were performed with both materials (see Section 6.4.11). Calculations in this chapter are therefore only performed for the Boral neutron absorber, with results directly applicable to Metamic.

The HI-STORM 100 System includes the HI-TRAC transfer cask and the HI-STORM storage cask. The HI-TRAC transfer cask is required for loading and unloading fuel into the MPC and for transfer of the MPC into the HI-STORM storage cask. HI-TRAC uses a lead shield for gamma radiation and a water-filled jacket for neutron shielding. The HI-STORM storage cask uses concrete as a shield for both gamma and neutron radiation. Both the HI-TRAC transfer cask and the HI-STORM storage cask, as well as the HI-STAR System†, accommodate the interchangeable MPC designs. The three cask designs (HI-STAR, HI-STORM, and HI-TRAC) differ only in the overpack reflector materials (steel for HI-STAR, concrete for HI-STORM, and lead for HI-TRAC), which do not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 system and vice versa. Therefore, the majority of criticality calculations to support both the HI-STAR and the HI-STORM System have been performed for only one of the two systems, namely the HI-STAR System. Only a selected number of analyses has been performed for both systems to demonstrate that this approach is valid. Therefore, unless specifically noted otherwise, all analyses documented throughout this chapter have been performed for the HI-STAR System. For the cases where analyses were performed for both the HI-STORM and HI-STAR System, this is clearly indicated.

The HI-STORM 100 System for storage (concrete overpack) is dry (no moderator), and thus, the reactivity is very low ($k_{eff} < 0.52$). However, the HI-STORM 100 System for cask transfer (HI-TRAC, lead overpack) is flooded for loading and unloading operations, and thus, represents the limiting case in terms of reactivity.

The MPC-24EF, MPC-32F and MPC-68FF contain the same basket as the MPC-24E, MPC-32 and MPC-68, respectively. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF, the MPC-32 and MPC-32F, and the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-24E, MPC-32 and MPC-68 are valid for the MPC-24EF, MPC-32F and MPC-68FF, respectively, and no separate analyses for the MPC-24EF, MPC-32F and MPC-68FF are necessary. Therefore, throughout this chapter and unless otherwise noted, ' MPC-68'refers to ' MPC-68 and/or MPC-68FF', ' MPC-24E' or ' MPC-24E/EF' refers to ' MPC-24Eand/or MPC-24EF', and ' MPC-32' or ' MPC-32/32F' refers to ' MPC-32 and/or MPC-32F'.

Confirmation of the criticality safety of the HI-STORM 100 System was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations

† Analyses for the HI-STAR System have previously been submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

were made with NITAWL-KENO5a from the SCALE-4.3 package [6.4.1]. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, and MCNP4a [6.1.4] were used. The CASMO-3 and MCNP4a calculations identify those tolerances that cause a positive reactivity effect, enabling the subsequent Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24~~,~~ *and* MPC-24E~~ and MPC-24EF~~) or cell spacing (MPC-32~~, MPC-32F~~, MPC-68~~, MPC-68F~~ and MPC-68F~~F~~), (3) the ^{10}B loading of the neutron absorber panels, and (4) the soluble boron concentration in the water. The critical experiment benchmarking is presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.

- 10CFR72.124, Criteria For Nuclear Criticality Safety.

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- Consistent with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.

- Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber.

- The fuel stack density is conservatively assumed to be at least 96% of theoretical (10.522 g/cm^3) for all criticality analyses. Fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels may be slightly greater than 96% of theoretical, the actual stack density will be less.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.

- When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.

- Neutron absorption in minor structural members and optional heat conduction elements is neglected, i.e., spacer grids, basket supports, and optional aluminum heat conduction elements are replaced by water.

- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.

- Planar-averaged enrichments are assumed for BWR fuel. (Consistent with NUREG-1536, analysis is presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)

- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000, consistent with NUREG-1536.

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

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

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

- For intact fuel assemblies, as defined in Table 1.0.1, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

Results of the design basis criticality safety calculations for single internally flooded HI-TRAC transfer casks with full water reflection on all sides (limiting cases for the HI-STORM 100 System), and for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System), loaded with intact fuel assemblies are listed in Tables 6.1.1 through 6.1.8, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. 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.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) loaded with damaged fuel assemblies or a combination of intact and damaged fuel assemblies are listed in Tables 6.1.9 through 6.1.12. The results include the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs

[††] 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.

qualified for damaged fuel and/or fuel debris (MPC-24E~~, MPC-24EF~~, MPC-68, MPC-68F~~, MPC-68FF, MPC-32~~ and MPC-32F), Tables 6.1.9 through 6.1.12 indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum k_{eff} value, the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration. For the permissible location of DFCs see Subsection 6.4.4.2. The maximum allowed enrichments are also listed in Section 2.1.9.

A table listing the maximum k_{eff} (including bias, uncertainties, and calculational statistics), calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum k_{eff} values for the HI-STORM 100 System are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

Table 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24
(no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	4.6	0.3080	0.9283	0.9296
14x14B	4.6	---	0.9237	0.9228
14x14C	4.6	---	0.9274	0.9287
14x14D	4.0	---	0.8531	0.8507
14x14E	5.0	---	0.7627	0.7627
15x15A	4.1	---	0.9205	0.9204
15x15B	4.1	---	0.9387	0.9388
15x15C	4.1	---	0.9362	0.9361
15x15D	4.1	---	0.9354	0.9367
15x15E	4.1	---	0.9392	0.9368
15x15F	4.1	0.3648	0.9393††	0.9395†††
15x15G	4.0	---	0.8878	0.8876
15x15H	3.8	---	0.9333	0.9337
16x16A	4.6	0.3588~~447~~	0.9322~~273~~	0.9327~~287~~
17x17A	4.0	0.3243	0.9378	0.9368
17x17B	4.0	---	0.9318	0.9324
17x17C	4.0	---	0.9319	0.9336

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

† 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.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9383.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9378.

Table 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24 WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**		
		HI-STORM	**HI-TRAC**	**HI-STAR**
14x14A	5.0	---	---	0.8884
14x14B	5.0	---	---	0.8900
14x14C	5.0	---	---	0.8950
14x14D	5.0	---	---	0.8518
14x14E	5.0	---	---	0.7132
15x15A	5.0	---	---	0.9119
15x15B	5.0	---	---	0.9284
15x15C	5.0	---	---	0.9236
15x15D	5.0	---	---	0.9261
15x15E	5.0	---	---	0.9265
15x15F	5.0	0.4013	0.9301	0.9314
15x15G	5.0	---	---	0.8939
15x15H	5.0	---	0.9345	0.9366
16x16A	5.0	---	---	0.8993~~55~~
17x17A	5.0	---	---	0.9264
17x17B	5.0	---	---	0.9284
17x17C	5.0	---	0.9296	0.9294

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

† 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 6.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E ~~AND MPC-24EF~~ (no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.9380
14x14B	5.0	---	---	0.9312
14x14C	5.0	---	---	0.9356
14x14D	5.0	---	---	0.8875
14x14E	5.0	---	---	0.7651
15x15A	4.5	---	---	0.9336
15x15B	4.5	---	---	0.9465
15x15C	4.5	---	---	0.9462
15x15D	4.5	---	---	0.9440
15x15E	4.5	---	---	0.9455
15x15F	4.5	0.3699	0.9465	0.9468
15x15G	4.5	---	---	0.9054
15x15H	4.2	---	---	0.9423
16x16A	5.0	---	---	0.9394~~41~~
17x17A	4.4	---	0.9467	0.9447
17x17B	4.4	---	---	0.9421
17x17C	4.4	---	---	0.9433

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

† 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 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E ~~AND MPC-24EF~~ WITH 300 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8963
14x14B	5.0	---	---	0.8974
14x14C	5.0	---	---	0.9031
14x14D	5.0	---	---	0.8588
14x14E	5.0	---	---	0.7249
15x15A	5.0	---	---	0.9161
15x15B	5.0	---	---	0.9321
15x15C	5.0	---	---	0.9271
15x15D	5.0	---	---	0.9290
15x15E	5.0	---	---	0.9309
15x15F	5.0	0.3897	0.9333	0.9332
15x15G	5.0	---	---	0.8972
15x15H	5.0	---	0.9399	0.9399
16x16A	5.0	---	---	0.90*60*~~21~~
17x17A	5.0	---	0.9320	0.9332
17x17B	5.0	---	---	0.9316
17x17C	5.0	---	---	0.9312

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

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

Table 6.1.5

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

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
16x16A	4.1	14~~3~~00	---	---	0.9375~~468~~
17x17A	4.1	1900	---	---	0.9111
17x17B	4.1	1900	---	---	0.9309
17x17C	4.1	1900	---	0.9365	0.9355

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

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

Table 6.1.6

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

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
16x16A	5.0	20~~1~~900	---	---	0.9429~~7~~4
17x17A	5.0	2600	---	---	0.9161
17x17B	5.0	2600	---	---	0.9371
17x17C	5.0	2600	---	0.9436	0.9437

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

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

Table 6.1.7

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 ~~AND MPC-68FF~~

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7††	---	0.7886	0.7888†††
6x6B‡	2.7††	---	0.7833	0.7824†††
6x6C	2.7††	0.2759	0.8024	0.8021†††
7x7A	2.7††	---	0.7963	0.7974†††
7x7B	4.2	0.4061	0.9385	0.9386
8x8A	2.7††	---	0.7690	0.7697†††
8x8B	4.2	0.3934	0.9427	0.9416
8x8C	4.2	0.3714	0.9429	0.9425
8x8D	4.2	---	0.9408	0.9403
8x8E	4.2	---	0.9309	0.9312
8x8F	4.0	---	0.9396	0.9411

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

† 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.

†† This calculation was performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} value is conservative.

††† This calculation was performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68 is at least 0.0310 g/cm^2. Therefore, the listed maximum k_{eff} value is conservative.

‡ Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the specification of authorized contents in Section 2.1.9.

Table 6.1.7 (continued)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 ~~AND MPC-68FF~~

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
9x9A	4.2	0.3365	0.9434	0.9417
9x9B	4.2	---	0.9417	0.9436
9x9C	4.2	---	0.9377	0.9395
9x9D	4.2	---	0.9387	0.9394
9x9E	4.0	.	0.9402	0.9401
9x9F	4.0	---	0.9402	0.9401
9x9G	4.2	---	0.9307	0.9309
10x10A	4.2	0.3379	0.9448‡‡	0.9457*
10x10B	4.2	---	0.9443	0.9436
10x10C	4.2	---	0.9430	0.9433
10x10D	4.0	---	0.9383	0.9376
10x10E	4.0	---	0.9157	0.9185

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

† 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.

‡‡ KENO5a verification calculation resulted in a maximum k_{eff} of 0.9451.

* KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

Table 6.1.8

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**		
		HI-STORM	**HI-TRAC**	**HI-STAR**
6x6A	2.7††	---	0.7886	0.7888
6x6B†††	2.7	---	0.7833	0.7824
6x6C	2.7	0.2759	0.8024	0.8021
7x7A	2.7	---	0.7963	0.7974
8x8A	2.7	---	0.7690	0.7697

Notes:

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

2. These calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

† 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.

†† These calculations were performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in the specification of authorized contents in Section 2.1.9.

Table 6.1.9

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-24E~~ AND MPC-24EF~~
WITH UP TO 4 DFCs

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)		Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}	
	Intact Fuel	Damaged Fuel and Fuel Debris		HI-TRAC	HI-STAR
All PWR Classes	4.0	4.0	0	0.9486	0.9480
All PWR Classes	5.0	5.0	600	0.9177	0.9185

Table 6.1.10

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68~~, MPC-68F~~ AND MPC-68FF
WITH UP TO 68 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)		Maximum k_{eff}	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
6x6A, 6x6B, 6x6C, 7x7A, 8x8A	2.7	2.7	0.8024	0.8021

Table 6.1.11

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

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)		Maximum k_{eff}	
	Intact Fuel	Damaged Fuel and Fuel Debris	HI-TRAC	HI-STAR
All BWR Classes	3.7	4.0	0.9328	0.9328

Table 6.1.12

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

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	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	4.1	1500	---	0.93*48*~~35~~
	5.0	2300	---	0.92*99*~~89~~
17x17A, B, C	4.1	2100	---	0.9294
	5.0	2900	---	0.9367

6.2 SPENT FUEL LOADING

Specifications for the BWR and PWR fuel assemblies that were analyzed are given in Tables 6.2.1 and 6.2.2, respectively. For the BWR fuel characteristics, the number and dimensions for the water rods are the actual number and dimensions. For the PWR fuel characteristics, the actual number and dimensions of the control rod guide tubes and thimbles are used. Table 6.2.1 lists 72 unique BWR assemblies while Table 6.2.2 lists 46 unique PWR assemblies, all of which were explicitly analyzed for this evaluation. Examination of Tables 6.2.1 and 6.2.2 reveals that there are a large number of minor variations in fuel assembly dimensions.

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in defining the authorized contents could limit the applicability of the HI-STORM 100 System. To resolve this limitation, bounding criticality analyses are presented in this section for a number of defined fuel assembly classes for both fuel types (PWR and BWR). The results of the bounding criticality analyses justify using bounding fuel dimensions for defining the authorized contents.

6.2.1 Definition of Assembly Classes

For each array size (e.g., 6x6, 7x7, 15x15, etc.), the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; (3) number and locations of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Tables 6.2.1 and 6.2.2, respectively. It should be noted that these assembly classes are unique to this evaluation and are not known to be consistent with any class designations in the open literature.

For each assembly class, calculations have been performed for all of the dimensional variations for which data is available (i.e., all data in Tables 6.2.1 and 6.2.2). These calculations demonstrate that the maximum reactivity corresponds to:

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

Therefore, for each assembly class, a bounding assembly was defined based on the above characteristics and a calculation for the bounding assembly was performed to demonstrate compliance with the regulatory requirement of k_{eff} < 0.95. In some assembly classes this bounding assembly corresponds directly to one of the actual (real) assemblies; while in most assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, the authorized contents in Section 2.1.9 are defined in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R). Additionally, parametric studies were performed for a PWR assembly (the 15x15F assembly class) in the MPC-24 and MPC-32 with soluble boron in the water flooding the MPC. The results of these studies are shown in Table 6.2.3 through 6.2.5, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod/guide tube thickness, (5) artificially replacing the Zircaloy water rod tubes/guide tubes with water, (6) maximizing the channel thickness (for BWR Assemblies), and (7) increasing the active length. These results, and the many that follow, justify the approach for using bounding dimensions for defining the authorized contents. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. As these studies were performed with and without soluble boron, they also demonstrate that the bounding dimensions are valid independent of the soluble boron concentration.

As mentioned, the bounding approach used in these analyses often results in a maximum k_{eff} value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

6.2.2 Intact PWR Fuel Assemblies

6.2.2.1 Intact PWR Fuel Assemblies in the MPC-24 without Soluble Boron

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum k_{eff} of 0.9395). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.20) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.6 through 6.2.22 for the fully flooded condition without soluble boron in the water.

Tables 6.2.6 through 6.2.22 show the maximum k_{eff} values for the assembly classes that are acceptable for storage in the MPC-24. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations for the MPC-24 were performed for a ^{10}B loading of 0.020 g/cm^2, which is 75% of the minimum loadingof 0.0267 g/cm^2 for Boral, or 90% of the minimum loading of 0.0223 g/cm^2 for Metamic. The maximum allowable enrichment in the MPC-24 varies from 3.8 to 5.0 wt% ^{235}U, depending on the assembly class, and is defined in Tables 6.2.6 through 6.2.22. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.6 through 6.2.22 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.6). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.9), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

The results of the analyses for the MPC-24, which were performed for all assemblies in each class (see Tables 6.2.6 through 6.2.22), further confirm the validity of the bounding dimensions established in Section 6.2.1. Thus, for all following calculations, namely analyses of the MPC-24E, MPC-32, and MPC-24 with soluble boron present in the water, only the bounding assembly in each class is analyzed.

6.2.2.2 <u>Intact PWR Fuel Assemblies in the MPC-24 with Soluble Boron</u>

Additionally, the HI-STAR 100 system is designed to allow credit for the soluble boron typically present in the water of PWR spent fuel pools. For a minimum soluble boron concentration of 400ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.2 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} of 0.9366). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.3 <u>Intact PWR Assemblies in the MPC-24E ~~and MPC-24EF~~ with and without Soluble Boron</u>

The MPC-24E ~~and MPC-24EF are~~*is a* variations of the MPC-24, which provide*s* for storage of higher enriched fuel than the MPC-24 through optimization of the storage cell layout. The MPC-24E ~~and MPC-24EF~~ also allow*s* for the loading of up to 4 PWR Damaged Fuel Containers

(DFC) with damaged PWR fuel ~~(MPC-24E and MPC-24EF)~~ and PWR fuel debris ~~(MPC-24EF only)~~. The requirements for damaged fuel and fuel debris in the MPC-24E ~~and MPC-24EF~~ *is*~~are~~ discussed in Section 6.2.4.3.

Without credit for soluble boron, the maximum allowable fuel enrichment varies between 4.2 and 5.0 wt% ^{235}U, depending on the assembly classes as identified in Tables 6.2.6 through 6.2.22. The maximum allowable enrichment for each assembly class is listed in Table 6.1.3, together with the maximum k_{eff} for the bounding assembly in the assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit The 15x15F assembly class at 4.5% enrichment has the highest reactivity (maximum k_{eff} of 0.9468). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a minimum soluble boron concentration of 300ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.4 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} of 0.9399). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.4 Intact PWR Assemblies in the MPC-32 ~~and MPC-32F~~

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

For a maximum allowable fuel enrichment of 4.1 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1300ppm and 1900ppm is required, depending on the assembly class. Table 6.1.5 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 16x16A assembly class has the highest reactivity (maximum k_{eff} of 0.9468). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a maximum allowable fuel enrichment of 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1900ppm and 2600ppm is required, depending on the assembly class. Table 6.1.6 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15F assembly class has the highest reactivity (maximum k_{eff} of 0.9483). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.3 Intact BWR Fuel Assemblies in the MPC-68~~-and MPC-68FF~~

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum k_{eff} of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 ~~and MPC-68FF~~ are summarized in Tables 6.2.23 through 6.2.40 for the fully flooded condition. In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.23 through 6.2.40 show the maximum k_{eff} values for assembly classes that are acceptable for storage in the MPC-68~~-and MPC-68FF~~. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 ~~and MPC-68FF~~ were performed with a ^{10}B loading of 0.0279 g/cm^2, which is 75% of the minimum loading of 0.0372 g/cm^2 for Boral, or 90% of the minimum loading of 0.031 g/cm^2 for Metamic. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a ^{10}B loading of 0.0067 g/cm^2. The maximum allowable enrichment in the MPC-68 ~~and MPC-68FF~~ varies from 2.7 to 4.2 wt% ^{235}U, depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.7 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68~~-and MPC-68FF~~.

Tables 6.2.23 through 6.2.40 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.27), the authorized contents dimensions are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain assemblies with thicker channels (e.g., 120 mils), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.23). All of the maximum k_{eff} values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum k_{eff} values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the specification of the authorized contents has no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum k_{eff} value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils thick; and (3) the active fuel length of the full length rods is 155 inches. Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

6.2.4 BWR and PWR Damaged Fuel Assemblies and Fuel Debris

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Table 1.0.1. Both damaged -fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. Five different DFC types with different cross sections are considered; three types for BWR fuel and two for PWR fuel. ~~DFCs containing fuel debris must be stored in the MPC-68F, MPC-68FF, MPC-24EF or MPC-32F.~~ DFCs containing BWR damaged fuel assemblies *and fuel debris* may be stored in the MPC-68 *or*~~,~~ MPC-68F~~-or MPC-68FF~~. DFCs containing PWR damaged fuel may be stored in the MPC-24E *or*~~,~~ ~~MPC-24EF,~~ MPC-32~~-or MPC-32F~~. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4.

6.2.4.1 Damaged BWR Fuel Assemblies and BWR Fuel Debris in Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

Tables 6.2.41 through 6.2.45 show the maximum k_{eff} values for the five assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum loading, 0.0089 g/cm^2. However, because the practical manufacturing lower limit for minimum ^{10}B loading is 0.01 g/cm^2, the minimum ^{10}B loading of 0.01 g/cm^2 is specified on the drawing in Section 1.5, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt% ^{235}U, while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt% ^{235}U in the specification of the authorized contents. Therefore, the maximum k_{eff} values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.41 through 6.2.45 for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt% ^{235}U enrichment has the highest reactivity (maximum k_{eff} of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual ^{10}B loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a ^{10}B loading of 0.0089 g/cm^2, which conservatively bounds the analysis of damaged BWR fuel assemblies in an MPC-68 ~~or MPC-68FF~~ with a minimum ^{10}B loading of 0.0372 g/cm^2, damaged BWR fuel assemblies *or fuel debris* may ~~also~~ be stored in the MPC-68 or MPC-68FF. ~~However, fuel debris is limited to the MPC-68F and MPC-68FF by the specification of the authorized contents.~~

Tables 6.2.41 through 6.2.45 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.43), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

6.2.4.2 Damaged BWR Fuel Assemblies and Fuel Debris in the MPC-68 ~~and MPC-68FF~~

Damaged BWR fuel assemblies and fuel debris from all BWR classes may be loaded into the MPC-68 ~~and MPC-68FF~~ by restricting the locations of the DFCs to 16 specific cells on the periphery of the fuel basket. The MPC-68 may be loaded with up to 16 DFCs containing damaged fuel assemblies, ~~. The MPC-68FF may also be loaded with up to 16 DFCs,~~ with up to 8 DFCs containing fuel debris.

For all assembly classes, the enrichment of the damaged fuel or fuel debris is limited to a maximum of 4.0 wt% ^{235}U, while the enrichment of the intact assemblies stored together with the damaged fuel is limited to a maximum of 3.7 wt% ^{235}U. The maximum k_{eff} is 0.9328. The criticality evaluation of the damaged fuel assemblies and fuel debris in the MPC-68 ~~and MPC-68FF~~ is presented in Section 6.4.4.2.

6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris

In addition to storing intact PWR fuel assemblies, the HI-STORM 100 System is designed to store damaged PWR fuel assemblies ~~(MPC-24E, MPC-24EF, MPC-32 and MPC-32F)~~ and fuel debris (MPC-24EF and MPC-32~~F only~~). 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 and MPC-24EF

Up to four DFCs may be stored in the MPC-24E~~-or MPC-24EF~~. When loaded with damaged fuel and/or fuel debris, the maximum enrichment for intact and damaged fuel is 4.0 wt% ^{235}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 and MPC-32F

Up to eight DFCs may be stored in the MPC-32~~-or MPC-32F~~. For a maximum allowable fuel enrichment of 4.1 wt% ^{235}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.

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~~-or MPC-68FF~~. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC is very low compared to the approved fuel assemblies (The ^{235}U content of these rods correspond to UO_2 rods with an initial enrichment of approximately 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 6.2.46. The criticality evaluation*s* are presented in Subsection 6.4.6.

Table 6.2.1 (page 1 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694	36	0.5645	0.0350	0.4940	110.0	0	n/a	n/a	0.060	4.290
6x6A02	Zr	0.694	36	0.5645	0.0360	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A03	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A04	Zr	0.694	36	0.5550	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A05	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A06	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6A07	Zr	0.700	36	0.5555	0.03525	0.4780	110.0	0	n/a	n/a	0.060	4.290
6x6A08	Zr	0.710	36	0.5625	0.0260	0.4980	110.0	0	n/a	n/a	0.060	4.290
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B02	Zr	0.694	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B03	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B04	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6B05	Zr	0.710	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6C Assembly Class												
6x6C01	Zr	0.740	36	0.5630	0.0320	0.4880	77.5	0	n/a	n/a	0.060	4.542
7x7A Assembly Class												
7x7A01	Zr	0.631	49	0.4860	0.0328	0.4110	80	0	n/a	n/a	0.060	4.542

Table 6.2.1 (page 2 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738	49	0.5630	0.0320	0.4870	150	0	n/a	n/a	0.080	5.278
7x7B02	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.102	5.291
7x7B03	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.080	5.278
7x7B04	Zr	0.738	49	0.5700	0.0355	0.4880	150	0	n/a	n/a	0.080	5.278
7x7B05	Zr	0.738	49	0.5630	0.0340	0.4775	150	0	n/a	n/a	0.080	5.278
7x7B06	Zr	0.738	49	0.5700	0.0355	0.4910	150	0	n/a	n/a	0.080	5.278
8x8A Assembly Class												
8x8A01	Zr	0.523	64	0.4120	0.0250	0.3580	110	0	n/a	n/a	0.100	4.290
8x8A02	Zr	0.523	63	0.4120	0.0250	0.3580	120	0	n/a	n/a	0.100	4.290

Table 6.2.1 (page 3 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B02	Zr	0.636	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B03	Zr	0.640	63	0.4930	0.0340	0.4160	150	1	0.493	0.425	0.100	5.278
8x8B04	Zr	0.642	64	0.5015	0.0360	0.4195	150	0	n/a	n/a	0.100	5.278
8x8C Assembly Class												
8x8C01	Zr	0.641	62	0.4840	0.0350	0.4050	150	2	0.484	0.414	0.100	5.278
8x8C02	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.000	no channel
8x8C03	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.080	5.278
8x8C04	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C05	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C06	Zr	0.640	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.100	5.278
8x8C07	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C08	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.493	0.425	0.100	5.278
8x8C09	Zr	0.640	62	0.4930	0.0340	0.4160	150	2	0.493	0.425	0.100	5.278
8x8C10	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C11	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.215
8x8C12	Zr	0.636	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.120	5.215

Table 6.2.1 (page 4 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640	60	0.4830	0.0320	0.4110	150	2 large/ 2 small	0.591/ 0.483	0.531/ 0.433	0.100	5.278
8x8D02	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.591	0.531	0.100	5.278
8x8D03	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.483	0.433	0.100	5.278
8x8D04	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.100	5.278
8x8D05	Zr	0.640	60	0.4830	0.0320	0.4100	150	1	1.34	1.26	0.100	5.278
8x8D06	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.120	5.278
8x8D07	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.080	5.278
8x8D08	Zr	0.640	61	0.4830	0.0300	0.4140	150	3	0.591	0.531	0.080	5.278
8x8E Assembly Class												
8x8E01	Zr	0.640	59	0.4930	0.0340	0.4160	150	5	0.493	0.425	0.100	5.278
8x8F Assembly Class												
8x8F01	Zr	0.609	64	0.4576	0.0290	0.3913	150	4†	0.291†	0.228†	0.055	5.390
9x9A Assembly Class												
9x9A01	Zr	0.566	74	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A02	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A03	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.100	5.278
9x9A04	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.120	5.278

† Four rectangular water cross segments dividing the assembly into four quadrants

Table 6.2.1 (page 5 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9B Assembly Class												
9x9B01	Zr	0.569	72	0.4330	0.0262	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B02	Zr	0.569	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B03	Zr	0.572	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9C Assembly Class												
9x9C01	Zr	0.572	80	0.4230	0.0295	0.3565	150	1	0.512	0.472	0.100	5.278
9x9D Assembly Class												
9x9D01	Zr	0.572	79	0.4240	0.0300	0.3565	150	2	0.424	0.364	0.100	5.278
9x9E Assembly Class†												
9x9E01	Zr	0.572	76	0.4170	0.0265	0.3530	150	5	0.546	0.522	0.120	5.215
9x9E02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215

† The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in Section 2.1.9.

Table 6.2.1 (page 6 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9F Assembly Class*												
9x9F01	Zr	0.572	76	0.4430	0.0285	0.3745	150	5	0.546	0.522	0.120	5.215
9x9F02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215
9x9G Assembly Class												
9x9G01	Zr	0.572	72	0.4240	0.0300	0.3565	150	1	1.668	1.604	0.120	5.278
10x10A Assembly Class												
10x10A01	Zr	0.510	92	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A02	Zr	0.510	78	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A03	Zr	0.510	92/78	0.4040	0.0260	0.3450	155/90	2	0.980	0.920	0.100	5.278
10x10B Assembly Class												
10x10B01	Zr	0.510	91	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B02	Zr	0.510	83	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B03	Zr	0.510	91/83	0.3957	0.0239	0.3413	155/90	1	1.378	1.321	0.100	5.278

* The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in Section 2.1.9.

Table 6.2.1 (page 7 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10C Assembly Class												
10x10C01	Zr	0.488	96	0.3780	0.0243	0.3224	150	5	1.227	1.165	0.055	5.347
10x10D Assembly Class												
10x10D01	SS	0.565	100	0.3960	0.0200	0.3500	83	0	n/a	n/a	0.08	5.663
10x10E Assembly Class												
10x10E01	SS	0.557	96	0.3940	0.0220	0.3430	83	4	0.3940	0.3500	0.08	5.663

Table 6.2.2 (page 1 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.527	0.493	0.0170
14x14A02	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.528	0.490	0.0190
14x14A03	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.526	0.492	0.0170
14x14B Assembly Class											
14x14B01	Zr	0.556	179	0.422	0.0243	0.3659	150	17	0.539	0.505	0.0170
14x14B02	Zr	0.556	179	0.417	0.0295	0.3505	150	17	0.541	0.507	0.0170
14x14B03	Zr	0.556	179	0.424	0.0300	0.3565	150	17	0.541	0.507	0.0170
14x14B04	Zr	0.556	179	0.426	0.0310	0.3565	150	17	0.541	0.507	0.0170
14x14C Assembly Class											
14x14C01	Zr	0.580	176	0.440	0.0280	0.3765	150	5	1.115	1.035	0.0400
14x14C02	Zr	0.580	176	0.440	0.0280	0.3770	150	5	1.115	1.035	0.0400
14x14C03	Zr	0.580	176	0.440	0.0260	0.3805	150	5	1.111	1.035	0.0380
14x14D Assembly Class											
14x14D01	SS	0.556	180	0.422	0.0165	0.3835	144	16	0.543	0.514	0.0145

Table 6.2.2 (page 2 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14E Assembly Class											
14x14E01†	SS	0.453 and 0.441	162 3 8	0.3415 0.3415 0.3415	0.0120 0.0285 0.0200	0.313 0.280 0.297	102	0	n/a	n/a	n/a
14x14E02†	SS	0.453 and 0.441	173	0.3415	0.0120	0.313	102	0	n/a	n/a	n/a
14x14E03†	SS	0.453 and 0.441	173	0.3415	0.0285	0.0280	102	0	n/a	n/a	n/a
15x15A Assembly Class											
15x15A01	Zr	0.550	204	0.418	0.0260	0.3580	150	21	0.533	0.500	0.0165

† This is the fuel assembly used at Indian Point 1 (IP-1). This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. It has a different pitch in different sections of the assembly, and different fuel rod dimensions in some rods.

Table 6.2.2 (page 3 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.533	0.499	0.0170
15x15B02	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.546	0.512	0.0170
15x15B03	Zr	0.563	204	0.422	0.0243	0.3660	150	21	0.533	0.499	0.0170
15x15B04	Zr	0.563	204	0.422	0.0243	0.3659	150	21	0.545	0.515	0.0150
15x15B05	Zr	0.563	204	0.422	0.0242	0.3659	150	21	0.545	0.515	0.0150
15x15B06	Zr	0.563	204	0.420	0.0240	0.3671	150	21	0.544	0.514	0.0150
15x15C Assembly Class											
15x15C01	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.493	0.0255
15x15C02	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.511	0.0165
15x15C03	Zr	0.563	204	0.424	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15C04	Zr	0.563	204	0.417	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15D Assembly Class											
15x15D01	Zr	0.568	208	0.430	0.0265	0.3690	150	17	0.530	0.498	0.0160
15x15D02	Zr	0.568	208	0.430	0.0265	0.3686	150	17	0.530	0.498	0.0160
15x15D03	Zr	0.568	208	0.430	0.0265	0.3700	150	17	0.530	0.499	0.0155
15x15D04	Zr	0.568	208	0.430	0.0250	0.3735	150	17	0.530	0.500	0.0150
15x15E Assembly Class											
15x15E01	Zr	0.568	208	0.428	0.0245	0.3707	150	17	0.528	0.500	0.0140
15x15F Assembly Class											
15x15F01	Zr	0.568	208	0.428	0.0230	0.3742	150	17	0.528	0.500	0.0140

Table 6.2.2 (page 4 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563	204	0.422	0.0165	0.3825	144	21	0.543	0.514	0.0145
15x15H Assembly Class											
15x15H01	Zr	0.568	208	0.414	0.0220	0.3622	150	17	0.528	0.500	0.0140
16x16A Assembly Class											
16x16A01	Zr	0.506	236	0.382	0.0250	0.3255	150	5	0.980	0.900	0.0400
16x16A02	Zr	0.506	236	0.382	0.0250	0.3250	150	5	0.980	0.900	0.0400
16x16A03	*Zr*	*0.506*	*236*	*0.382*	*0.0235*	*0.3255*	*150*	*5*	*0.970*	*0.900*	*0.0350*
17x17A Assembly Class											
17x17A01	Zr	0.496	264	0.360	0.0225	0.3088	150	25	0.474	0.442	0.0160
17x17A02	Zr	0.496	264	0.360	0.0250	0.3030	150	25	0.480	0.448	0.0160
17x17B Assembly Class											
17x17B01	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.482	0.450	0.0160
17x17B02	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.474	0.442	0.0160
17x17B03	Zr	0.496	264	0.376	0.0240	0.3215	150	25	0.480	0.448	0.0160
17x17B04	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.427	0.399	0.0140
17x17B05	Zr	0.496	264	0.374	0.0240	0.3195	150	25	0.482	0.450	0.0160
17x17B06	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.480	0.452	0.0140
17x17C Assembly Class											
17x17C01	Zr	0.502	264	0.379	0.0240	0.3232	150	25	0.472	0.432	0.0200
17x17C02	Zr	0.502	264	0.377	0.0220	0.3252	150	25	0.472	0.432	0.0200

Table 6.2.3
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-68
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	water rod thickness	channel thickness
8x8C04 (GE8x8R)	reference	0.9307	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
increase pellet OD (+0.001)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.411	0.030	0.100
decrease pellet OD (-0.001)	-0.0008	0.9299	0.0009	0.483	0.419	0.032	0.409	0.030	0.100
increase clad ID (+0.004)	+0.0027	0.9334	0.0007	0.483	0.423	0.030	0.410	0.030	0.100
decrease clad ID (-0.004)	-0.0034	0.9273	0.0007	0.483	0.415	0.034	0.410	0.030	0.100
increase clad OD (+0.004)	-0.0041	0.9266	0.0008	0.487	0.419	0.034	0.410	0.030	0.100
decrease clad OD (-0.004)	+0.0023	0.9330	0.0007	0.479	0.419	0.030	0.410	0.030	0.100
increase water rod thickness (+0.015)	-0.0019	0.9288	0.0008	0.483	0.419	0.032	0.410	0.045	0.100
decrease water rod thickness (-0.015)	+0.0001	0.9308	0.0008	0.483	0.419	0.032	0.410	0.015	0.100
remove water rods (i.e., replace the water rod tubes with water)	+0.0021	0.9328	0.0008	0.483	0.419	0.032	0.410	0.000	0.100
remove channel	-0.0039	0.9268	0.0009	0.483	0.419	0.032	0.410	0.030	0.000
increase channel thickness (+0.020)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.410	0.030	0.120
reduced active length (120 Inches)	-0.0007	0.9300	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
reduced active length (90 Inches)	-0.0043	0.9264	0.0007	0.483	0.419	0.032	0.410	0.030	0.100

Table 6.2.4
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC 24 with 400ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9271	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	-0.0008	0.9263	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	-0.0002	0.9269	0.0005	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0040	0.9311	0.0005	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0033	0.9238	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0042	0.9229	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0035	0.9306	0.0005	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0008	0.9263	0.0005	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	+0.0006	0.9277	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	+0.0028	0.9299	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	-0.0318	0.8953	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2.5
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-32 with 2600ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9389	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	+0.0019	0.9408	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	0.0000	0.9389	0.0004	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0015	0.9404	0.0004	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0015	0.9374	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0002	0.9387	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0007	0.9397	0.0004	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0003	0.9387	0.0004	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	-0.0005	0.9384	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	-0.0005	0.9385	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	+0.0039	0.9428	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2.6
MAXIMUM K_{EFF} VALUES FOR THE 14X14A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14A (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14A01	0.9295	0.9252	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
14x14A02	0.9286	0.9242	0.0008	0.400	0.3514	0.0243	0.3444	150	0.019
14x14A03	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
Dimensions Listed for Authorized Contents				0.400 (min.)	0.3514 (max.)		0.3444 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (14x14A03)	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017

Table 6.2.7
MAXIMUM K_{EFF} VALUES FOR THE 14X14B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14B (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14B01	0.9159	0.9117	0.0007	0.422	0.3734	0.0243	0.3659	150	0.017
14x14B02	0.9169	0.9126	0.0008	0.417	0.3580	0.0295	0.3505	150	0.017
14x14B03	0.9110	0.9065	0.0009	0.424	0.3640	0.0300	0.3565	150	0.017
14x14B04	0.9084	0.9039	0.0009	0.426	0.3640	0.0310	0.3565	150	0.017
Dimensions Listed for Authorized Contents				0.417 (min.)	0.3734 (max.)		0.3659 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (B14x14B01)	0.9228	0.9185	0.0008	0.417	0.3734	0.0218	0.3659	150	0.017

Table 6.2.8
MAXIMUM K_{EFF} VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14C (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 176 fuel rods, 5 guide tubes, pitch=0.580, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9258	0.9215	0.0008	0.440	0.3840	0.0280	0.3765	150	0.040
14x14C02	0.9265	0.9222	0.0008	0.440	0.3840	0.0280	0.3770	150	0.040
14x14C03	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038
Dimensions Listed for Authorized Contents				0.440 (min.)	0.3880 (max.)		0.3805 (max.)	150 (max.)	0.038 (min.)
bounding dimensions (14x14C03)	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038

Table 6.2.9
MAXIMUM K_{EFF} VALUES FOR THE 14X14D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14D (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 180 fuel rods, 16 guide tubes, pitch=0.556, SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14D01	0.8507	0.8464	0.0008	0.422	0.3890	0.0165	0.3835	144	0.0145
Dimensions Listed for Authorized Contents				0.422 (min.)	0.3890 (max.)		0.3835 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.10
MAXIMUM K_{EFF} VALUES FOR THE 14X14E ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14E (5.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 173 fuel rods, 0 guide tubes, pitch=0.453 and 0.441, SS clad†									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length††	guide tube thickness
14x14E01	0.7598	0.7555	0.0008	0.3415	0.3175 0.2845 0.3015	0.0120 0.0285 0.0200	0.3130 0.2800 0.2970	102	0.0000
14x14E02	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000
14x14E03	0.6952	0.6909	0.0008	0.3415	0.2845	0.0285	0.2800	102	0.0000
Dimensions Listed for Authorized Contents				0.3415 (min.)	0.3175 (max.)		0.3130 (max.)	102 (max.)	0.0000 (min.)
Bounding dimensions (14x14E02)	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000

† This is the IP-1 fuel assembly at Indian Point. This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. Fuel rod dimensions are bounding for each of the three types of rods found in the IP-1 fuel assembly.

†† Calculations were conservatively performed for a fuel length of 150 inches.

Table 6.2.11
MAXIMUM K_{EFF} VALUES FOR THE 15X15A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15A (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.550, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15A01	0.9204	0.9159	0.0009	0.418	0.3660	0.0260	0.3580	150	0.0165
Dimensions Listed for Authorized Contents				0.418 (min.)	0.3660 (max.)		0.3580 (max.)	150 (max.)	0.0165 (min.)

Table 6.2.12
MAXIMUM K_{EFF} VALUES FOR THE 15X15B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15B (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15B01	0.9369	0.9326	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B02	0.9338	0.9295	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B03	0.9362	0.9318	0.0008	0.422	0.3734	0.0243	0.3660	150	0.017
15x15B04	0.9370	0.9327	0.0008	0.422	0.3734	0.0243	0.3659	150	0.015
15x15B05	0.9356	0.9313	0.0008	0.422	0.3736	0.0242	0.3659	150	0.015
15x15B06	0.9366	0.9324	0.0007	0.420	0.3720	0.0240	0.3671	150	0.015
Dimensions Listed for Authorized Contents				0.420 (min.)	0.3736 (max.)		0.3671 (max.)	150 (max.)	0.015 (min.)
bounding dimensions (B15x15B01)	0.9388	0.9343	0.0009	0.420	0.3736	0.0232	0.3671	150	0.015

Table 6.2.13
MAXIMUM K_{EFF} VALUES FOR THE 15X15C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15C (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15C01	0.9255	0.9213	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0255
15x15C02	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0165
15x15C03	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3565	150	0.0165
15x15C04	0.9311	0.9268	0.0008	0.417	0.3570	0.0300	0.3565	150	0.0165
Dimensions Listed for Authorized Contents				0.417 (min.)	0.3640 (max.)		0.3570 (max.)	150 (max.)	0.0165 (min.)
bounding dimensions (B15x15C01)	0.9361	0.9316	0.0009	0.417	0.3640	0.0265	0.3570	150	0.0165

Table 6.2.14
MAXIMUM K_{EFF} VALUES FOR THE 15X15D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15D (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15D01	0.9341	0.9298	0.0008	0.430	0.3770	0.0265	0.3690	150	0.0160
15x15D02	0.9367	0.9324	0.0008	0.430	0.3770	0.0265	0.3686	150	0.0160
15x15D03	0.9354	0.9311	0.0008	0.430	0.3770	0.0265	0.3700	150	0.0155
15x15D04	0.9339	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150
Dimensions Listed for Authorized Contents				0.430 (min.)	0.3800 (max.)		0.3735 (max.)	150 (max.)	0.0150 (min.)
bounding dimensions (15x15D04)	0.9339†	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150

† The k_{eff} value listed for the 15x15D02 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9367 (15x15D02) value is listed in Table 6.1.1 as the maximum.

Table 6.2.15
MAXIMUM K_{EFF} VALUES FOR THE 15X15E ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15E (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15E01	0.9368	0.9325	0.0008	0.428	0.3790	0.0245	0.3707	150	0.0140
Dimensions Listed for Authorized Contents				0.428 (min.)	0.3790 (max.)		0.3707 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.16
MAXIMUM K_{EFF} VALUES FOR THE 15X15F ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15F (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15F01	0.9395†	0.9350	0.0009	0.428	0.3820	0.0230	0.3742	150	0.0140
Dimensions Listed for Authorized Contents				0.428 (min.)	0.3820 (max.)		0.3742 (max.)	150 (max.)	0.0140 (min.)

† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9383.

Table 6.2.17
MAXIMUM K_{EFF} VALUES FOR THE 15X15G ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15G (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15G01	0.8876	0.8833	0.0008	0.422	0.3890	0.0165	0.3825	144	0.0145
Dimensions Listed for Authorized Contents				0.422 (min.)	0.3890 (max.)		0.3825 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.18
MAXIMUM K_{EFF} VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15H (3.8% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9337	0.9292	0.0009	0.414	0.3700	0.0220	0.3622	150	0.0140
Dimensions Listed for Authorized Contents				0.414 (min.)	0.3700 (max.)		0.3622 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.19
MAXIMUM K_{EFF} VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

16x16A (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400
16x16A02	0.9263	0.9221	0.0007	0.382	0.3320	0.0250	0.3250	150	0.0400
16x16A03	*0.9327*	*0.9282*	*0.0009*	*0.382*	*0.3350*	*0.0235*	*0.3255*	*150*	*0.0350*
Dimensions Listed for Authorized Contents				0.382 (min.)	0.33*5*~~2~~0 (max.)		0.3255 (max.)	150 (max.)	0.0*350*~~400~~ (min.)
bounding dimensions (16x16A0*3*~~1~~)	0.9*327*~~287~~	0.9*282*~~244~~	0.0009~~8~~	0.382	0.33*5*~~2~~0	0.02*35*~~50~~	0.3255	150	0.0*350*~~400~~

Table 6.2.20
MAXIMUM K_{EFF} VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17A (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17A01	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016
17x17A02	0.9329	0.9286	0.0008	0.360	0.3100	0.0250	0.3030	150	0.016
Dimensions Listed for Authorized Contents				0.360 (min.)	0.3150 (max.)		0.3088 (max.)	150 (max.)	0.016 (min.)
bounding dimensions (17x17A01)	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016

Table 6.2.21
MAXIMUM K_{EFF} VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17B (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9288	0.9243	0.0009	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B02	0.9290	0.9247	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B03	0.9243	0.9199	0.0008	0.376	0.3280	0.0240	0.3215	150	0.016
17x17B04	0.9324	0.9279	0.0009	0.372	0.3310	0.0205	0.3232	150	0.014
17x17B05	0.9266	0.9222	0.0008	0.374	0.3260	0.0240	0.3195	150	0.016
17x17B06	0.9311	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014
Dimensions Listed for Authorized Contents				0.372 (min.)	0.3310 (max.)		0.3232 (max.)	150 (max.)	0.014 (min.)
bounding dimensions (17x17B06)	0.9311†	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014

† The k_{eff} value listed for the 17x17B04 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9324 (17x17B04) value is listed in Table 6.1.1 as the maximum.

Table 6.2.22
MAXIMUM K_{EFF} VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17C (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.502, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9293	0.9250	0.0008	0.379	0.3310	0.0240	0.3232	150	0.020
17x17C02	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020
Dimensions Listed for Authorized Contents				0.377 (min.)	0.3330 (max.)		0.3252 (max.)	150 (max.)	0.020 (min.)
bounding dimensions (17x17C02)	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020

Table 6.2.23
MAXIMUM K_{EFF} VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~
(all dimensions are in inches)

7x7B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.738, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630	0.4990	0.0320	0.4870	150	n/a	0.080
7x7B02	0.9301	0.9260	0.0007	0.5630	0.4890	0.0370	0.4770	150	n/a	0.102
7x7B03	0.9313	0.9271	0.0008	0.5630	0.4890	0.0370	0.4770	150	n/a	0.080
7x7B04	0.9311	0.9270	0.0007	0.5700	0.4990	0.0355	0.4880	150	n/a	0.080
7x7B05	0.9350	0.9306	0.0008	0.5630	0.4950	0.0340	0.4775	150	n/a	0.080
7x7B06	0.9298	0.9260	0.0006	0.5700	0.4990	0.0355	0.4910	150	n/a	0.080
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4910 (max.)	150 (max.)	n/a	0.120 (max.)
bounding dimensions (B7x7B01)	0.9375	0.9332	0.0008	0.5630	0.4990	0.0320	0.4910	150	n/a	0.102
bounding dimensions with 120 mil channel (B7x7B02)	0.9386	0.9344	0.0007	0.5630	0.4990	0.0320	0.4910	150	n/a	0.120

Table 6.2.24
MAXIMUM K_{EFF} VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68~~-and MPC-68FF~~
(all dimensions are in inches)

8x8B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 63 or 64 fuel rods†, 1 or 0 water rods†, pitch† = 0.636-0.642, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Fuel rods	Pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	63	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B02	0.9227	0.9185	0.0007	63	0.636	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B03	0.9299	0.9257	0.0008	63	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8B04	0.9236	0.9194	0.0008	64	0.642	0.5015	0.4295	0.0360	0.4195	150	n/a	0.100
Dimensions Listed for Authorized Contents				63 or 64	0.636-0.642	0.4840 (min.)	0.4295 (max.)		0.4195 (max.)	150 (max.)	0.034	0.120 (max.)
bounding (pitch=0.636) (B8x8B01)	0.9346	0.9301	0.0009	63	0.636	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.640) (B8x8B02)	0.9385	0.9343	0.0008	63	0.640	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.642) (B8x8B03)	0.9416	0.9375	0.0007	63	0.642	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.25

MAXIMUM K_{EFF} VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68~~-and MPC-68FF~~

(all dimensions are in inches)

8x8C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 62 fuel rods, 2 water rods, pitch† = 0.636-0.641, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8C02	0.9313	0.9268	0.0009	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.000
8x8C03	0.9329	0.9286	0.0008	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.800
8x8C04	0.9348††	0.9307	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.100
8x8C05	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.120
8x8C06	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8C07	0.9314	0.9273	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.100
8x8C08	0.9339	0.9298	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.034	0.100
8x8C09	0.9301	0.9260	0.0007	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8C10	0.9317	0.9275	0.0008	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C11	0.9328	0.9287	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C12	0.9285	0.9242	0.0008	0.636	0.4830	0.4190	0.0320	0.4110	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.636-0.641	0.4830 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
Bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9343.

Table 6.2.26

MAXIMUM K_{EFF} VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~

(all dimensions are in inches)

8x8D (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 60-61 fuel rods, 1-4 water rods†, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830	0.4190	0.0320	0.4110	150	0.03/0.025	0.100
8x8D02	0.9325	0.9284	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8D03	0.9351	0.9309	0.0008	0.4830	0.4190	0.0320	0.4110	150	0.025	0.100
8x8D04	0.9338	0.9296	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.100
8x8D05	0.9339	0.9294	0.0009	0.4830	0.4190	0.0320	0.4100	150	0.040	0.100
8x8D06	0.9365	0.9324	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.120
8x8D07	0.9341	0.9297	0.0009	0.4830	0.4190	0.0320	0.4110	150	0.040	0.080
8x8D08	0.9376	0.9332	0.0009	0.4830	0.4230	0.0300	0.4140	150	0.030	0.080
Dimensions Listed for Authorized Contents				0.4830 (min.)	0.4230 (max.)		0.4140 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B8x8D01)	0.9403	0.9363	0.0007	0.4830	0.4230	0.0300	0.4140	150	0.000	0.120

† Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.

Table 6.2.27
MAXIMUM K_{EFF} VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~
(all dimensions are in inches)

8x8E (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 59 fuel rods, 5 water rods, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
Dimensions Listed for Authorized Contents				0.4930 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.034 (min.)	0.100 (max.)

Table 6.2.28

MAXIMUM K_{EFF} VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~

(all dimensions are in inches)

8x8F (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9411	0.9366	0.0009	0.4576	0.3996	0.0290	0.3913	150	0.0315	0.055
Dimensions Listed for Authorized Contents				0.4576 (min.)	0.3996 (max.)		0.3913 (max.)	150 (max.)	0.0315 (min.)	0.055 (max.)

Table 6.2.29
MAXIMUM K_{EFF} VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~
(all dimensions are in inches)

9x9A (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 74/66 fuel rods†, 2 water rods, pitch=0.566, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400	0.3840	0.0280	0.3760	150/90	0.030	0.100
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400	0.3840	0.0280	0.3760	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.4400 (min.)	0.3840 (max.)		0.3760 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.000	0.120

† This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

Table 6.2.30
MAXIMUM K_{EFF} VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~
(all dimensions are in inches)

9x9B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 to 0.572†, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9380	0.9336	0.0008	0.569	0.4330	0.3807	0.0262	0.3737	150	0.0285	0.100
9x9B02	0.9373	0.9329	0.0009	0.569	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
9x9B03	0.9417	0.9374	0.0008	0.572	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
Dimensions Listed for Authorized Contents				0.572	0.4330 (min.)	0.3810 (max.)		0.3740 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B9x9B01)	0.9436	0.9394	0.0008	0.572	0.4330	0.3810	0.0260	0.3740††	150	0.000	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch.
†† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.31
MAXIMUM K_{EFF} VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~
(all dimensions are in inches)

9x9C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 80 fuel rods, 1 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230	0.3640	0.0295	0.3565	150	0.020	0.100
Dimensions Listed for Authorized Contents				0.4230 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.020 (min.)	0.100 (max.)

Table 6.2.32
MAXIMUM K_{EFF} VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~
(all dimensions are in inches)

9x9D (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 79 fuel rods, 2 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.9394	0.9350	0.0009	0.4240	0.3640	0.0300	0.3565	150	0.0300	0.100
Dimensions Listed for Authorized Contents				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0300 (min.)	0.100 (max.)

Table 6.2.33
MAXIMUM K_{EFF} VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~

(all dimensions are in inches)

9x9E (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.9334	0.9293	0.0007	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
9x9E02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents†				0.4170 (min.)	0.3640 (max.)		0.3530 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9E02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.34

MAXIMUM K_{EFF} VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~

(all dimensions are in inches)

9x9F (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2)

76 fuel rods, 5 water rods, pitch=0.572, Zr clad

Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.9307	0.9265	0.0007	0.4430	0.3860	0.0285	0.3745	150	0.0120	0.120
9x9F02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed for Authorized Contents†				0.4430 (min.)	0.3860 (max.)		0.3745 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9F02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for the authorized contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Contents lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.35

MAXIMUM K_{EFF} VALUES FOR THE 9X9G ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~

(all dimensions are in inches)

9x9G (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9G01	0.9309	0.9265	0.0008	0.4240	0.3640	0.0300	0.3565	150	0.0320	0.120
Dimensions Listed for Authorized Contents				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0320 (min.)	0.120 (max.)

Table 6.2.36
MAXIMUM K_{EFF} VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~

(all dimensions are in inches)

10x10A (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 92/78 fuel rods†, 2 water rods, pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040	0.3520	0.0260	0.3450	155/90	0.030	0.100
Dimensions Listed for Authorized Contents				0.4040 (min.)	0.3520 (max.)		0.3455 (max.)	150†† (max.)	0.030 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457†††	0.9414	0.0008	0.4040	0.3520	0.0260	0.3455‡	155	0.030	0.120

† This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for the authorized contents limits the active fuel length to 150 inches. This is due to the fact that the fixed neutron absorber panels are 156 inches in length.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

‡ This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.37

MAXIMUM K_{EFF} VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~

(all dimensions are in inches)

10x10B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 91/83 fuel rods†, 1 water rods (square, replacing 9 fuel rods), pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957	0.3480	0.0239	0.3413	155/90	0.0285	0.100
Dimensions Listed for Authorized Contents				0.3957 (min.)	0.3480 (max.)		0.3420 (max.)	150†† (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957	0.3480	0.0239	0.3420†††	155	0.000	0.120

† This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for the authorized contents limits the active fuel length to 150 inches. This is due to the fact that the fixed neutron absorber panels are 156 inches in length.

††† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.38
MAXIMUM K_{EFF} VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~

(all dimensions are in inches)

10x10C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.9433	0.9392	0.0007	0.3780	0.3294	0.0243	0.3224	150	0.031	0.055
Dimensions Listed for Authorized Contents				0.3780 (min.)	0.3294 (max.)		0.3224 (max.)	150 (max.)	0.031 (min.)	0.055 (max.)

Table 6.2.39

MAXIMUM K_{EFF} VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68 ~~and MPC-68FF~~

(all dimensions are in inches)

10x10D (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 100 fuel rods, 0 water rods, pitch=0.565, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960	0.3560	0.0200	0.350	83	n/a	0.080
Dimensions Listed for Authorized Contents				0.3960 (min.)	0.3560 (max.)		0.350 (max.)	83 (max.)	n/a	0.080 (max.)

Table 6.2.40
MAXIMUM K_{EFF} VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68~~ and MPC-68FF~~

(all dimensions are in inches)

10x10E (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 4 water rods, pitch=0.557, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940	0.3500	0.0220	0.3430	83	0.022	0.080
Dimensions Listed for Authorized Contents				0.3940 (min.)	0.3500 (max.)		0.3430 (max.)	83 (max.)	0.022 (min.)	0.080 (max.)

Table 6.2.41
MAXIMUM K_{EFF} VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F ~~and MPC-68FF~~
(all dimensions are in inches)

6x6A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 35 or 36 fuel rods††, 1 or 0 water rods††, pitch††=0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4940	110	n/a	0.060
6x6A02	0.7517	0.7476	0.0007	0.694	36	0.5645	0.4925	0.0360	0.4820	110	n/a	0.060
6x6A03	0.7545	0.7501	0.0008	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6A04	0.7537	0.7494	0.0008	0.694	36	0.5550	0.4850	0.0350	0.4820	110	n/a	0.060
6x6A05	0.7555	0.7512	0.0008	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6A06	0.7618	0.7576	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6A07	0.7588	0.7550	0.0005	0.700	36	0.5555	0.4850	0.03525	0.4780	110	n/a	0.060
6x6A08	0.7808	0.7766	0.0007	0.710	36	0.5625	0.5105	0.0260	0.4980	110	n/a	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5550 (min.)	0.5105 (max.)	0.02225	0.4980 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6A01)	0.7727	0.7685	0.0007	0.694	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A02)	0.7782	0.7738	0.0008	0.700	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A03)	0.7888	0.7846	0.0007	0.710	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.42
MAXIMUM K_{EFF} VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F~~and MPC-68FF~~
(all dimensions are in inches)

6x6B (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2)

35 or 36 fuel rods†† (up to 9 MOX rods), 1 or 0 water rods††, pitch††=0.694 to 0.710, Zr clad

Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	0.7604	0.7563	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6B02	0.7618	0.7577	0.0007	0.694	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B03	0.7619	0.7578	0.0007	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B04	0.7686	0.7644	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6B05	0.7824	0.7785	0.0006	0.710	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
Dimensions Listed for Authorized Contents				0.710 (max.)	35 or 36	0.5625 (min.)	0.4945 (max.)		0.4820 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6B01)	0.7822†††	0.7783	0.0006	0.710	35	0.5625	0.4945	0.0340	0.4820	120	0.0	0.060

Note:

1. These assemblies contain up to 9 MOX pins. The composition of the MOX fuel pins is given in Table 6.3.4.

† The ^{235}U enrichment of the MOX and UO_2 pins is assumed to be 0.711% and 3.0%, respectively.

†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

††† The k_{eff} value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1σ). Therefore, the 0.7824 value is listed in Tables 6.1.7 and 6.1.8 as the maximum.

Table 6.2.43
MAXIMUM K_{EFF} VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F ~~and MPC-68FF~~

(all dimensions are in inches)

6x6C (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 36 fuel rods, 0 water rods, pitch=0.740, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630	0.4990	0.0320	0.4880	77.5	n/a	0.060
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4880 (max.)	77.5 (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

Table 6.2.44

MAXIMUM K_{EFF} VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F ~~and MPC-68FF~~

(all dimensions are in inches)

7x7A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2)										
49 fuel rods, 0 water rods, pitch=0.631, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.7974	0.7932	0.0008	0.4860	0.4204	0.0328	0.4110	80	n/a	0.060
Dimensions Listed for Authorized Contents				0.4860 (min.)	0.4204 (max.)		0.4110 (max.)	80 (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.

Table 6.2.45
MAXIMUM K_{EFF} VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F ~~and MPC-68FF~~

(all dimensions are in inches)

8x8A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 63 or 64 fuel rods††, 0 water rods, pitch=0.523, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	64	0.4120	0.3620	0.0250	0.3580	110	n/a	0.100
8x8A02	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100
Dimensions Listed for Authorized Contents				63	0.4120 (min.)	0.3620 (max.)		0.3580 (max.)	120 (max.)	n/a	0.100 (max.)
bounding dimensions (8x8A02)	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for the authorized contents to 2.7%.
†† This assembly class was analyzed and qualified for a variation in the number of fuel rods.

Table 6.2.46

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	4.81"
Canister Wall Thickness	0.11"
Separator Assembly Plates Thickness	0.11"
Cladding OD	0.412"
Cladding ID	0.362"
Pellet OD	0.358"
Active Length	110.5"
Fuel Composition	1.8% UO_2 and 98.2% 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

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1, 6.3.1.a, 6.3.2 and 6.3.3 show representative horizontal cross sections of the four types of cells used in the calculations, and Figures 6.3.4 through 6.3.6 illustrate the basket configurations used. Four different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket
- an optimized 24 PWR assembly basket (24E~~/24EF~~)
- a 32 PWR assembly basket
- a 68 BWR assembly basket.

For all four basket designs, the same techniques and the same level of detail are used in the calculational models.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7. Although the fixed neutron absorber panels are 156 inches in length, which is much longer than the active fuel length (maximum of 150 inches), they are assumed equal to or less than the active fuel length in the calculations. As shown on the Drawings in Section 1.5, 16 of the 24 periphery fixed neutron absorber panels on the MPC-24 and MPC-24E~~/EF~~ have reduced width (i.e., 6.25 inches wide as opposed to 7.5 inches). However, the calculational models for these baskets conservatively assume all of the periphery fixed neutron absorber panels are 6.25 inches in width. Note that Figures 6.3.1 through 6.3.3 show Boral as the fixed neutron absorber. The effect of using Metamic as fixed neutron absorber is discussed in Subsection 6.4.11.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design conditions important to criticality safety (see Subsection 6.4.2.5), and thus from a criticality standpoint, the normal, off-normal, and accident conditions are identical and do not require individual models.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and fixed neutron absorber panels on the stainless steel walls of the storage cells. Under the conditions of storage, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ($k_{eff} < 0.52$). For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel rod pellet-to-clad gaps. Appendix 6.D provides sample input files for two of the MPC basket designs (MPC-68 and MPC-24) in the HI-STORM 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included. Furthermore, the water above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

As indicated in Figures 6.3.1 through 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO-3 and MCNP4a were used to determine the direction of the manufacturing tolerances, which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity.

CASMO-3 was used for one of each of the two principal basket designs, i.e. for the flux trap design MPC-24 and for the non-fluxtrap design MPC-68. The effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. Generally, the conclusions in Table 6.3.1 are directly applicable to the MPC-24E/~~EF~~ and the MPC-32. Exceptions are the conclusions for the water temperature and void percentage, which are not directly applicable to the MPC-32 due to the presence of high soluble boron concentrations in this canister. This condition is addressed in Section 6.4.2.1 where the optimum moderation is determined for the MPC-32.

Additionally, MCNP4a calculations are performed to evaluate the tolerances of the various basket dimensions of the MPC-68, MPC-24 and MPC-32 in further detail. The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a. Calculated k_{eff} results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR assemblies. Each of the basket dimensions are evaluated for their minimum, nominal and maximum values from the Drawings of section 1.5. For PWR MPC designs, the reactivity effect of tolerances with soluble boron present in the water is additionally determined. Due to the close similarity between the MPC-24 and MPC-24E, the basket dimensions are only evaluated for the MPC-24, and the same dimensional assumptions are applied to both MPC designs.

Based on the MCNP4a and CASMO-3 calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined. Because the reactivity effect (positive or negative) of the manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

As demonstrated in this section, design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, the fixed neutron absorbing panels and the soluble boron concentration in the water during loading/unloading operations. As shown in Chapter 11, none of these parameters are affected during any of the design basis off-normal or accident conditions involving handling, packaging, transfer or storage.

The MPC-32 criticality model uses a sheathing thickness of 0.075 inches, whereas the actual MPC-32 design uses a sheathing thickness of 0.035 inches. For the minimum cell pitch of 9.158 inches, the thicker sheathing results in a slightly smaller cell ID of 8.69 inches (minimum), compared to 8.73 inches (minimum) for the thinner sheathing. To demonstrate that the dimensions used in the criticality model are acceptable and conservative, calculations were performed for both sheathing thicknesses and the results are compared in Table 6.3.5. To bound various soluble boron levels, two comparisons were performed. The first comparison uses the bounding case for the MPC-32 (see Table 6.1.6), which is for assembly class 15x15F at 5 wt% ^{235}U and a soluble boron level of 2600 ppm. To bound lower soluble boron levels, the second comparison uses the same assembly class (15x15F), 0 ppm soluble boron (i.e. pure water), and an arbitrary enrichment of 1.7 wt% ^{235}U. In both comparisons, the results of the 0.075 inch sheathing are slightly higher, i.e. more conservative, than the results for 0.035 inch sheathing, although the differences are within the statistical uncertainties. Using a sheathing thickness of 0.075 inches in the criticality models of the MPC-32 is therefore acceptable, and potentially more conservative, than using the actual value of 0.035 inches. This validates the choice of the dimensional assumptions for the MPC-32 shown in Table 6.3.3, which are used for all further MPC-32 criticality calculations, unless otherwise noted.

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM 100 System are listed in Table 6.3.4.

The HI-STORM 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of fixed neutron absorber are provided in Section 1.2.1.3.1.

The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Section 9.1.5.3, to validate the ^{10}B (poison) concentration in the fixed neutron absorber. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an MCNP4a calculation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of ^{10}B atoms destroyed is only 2.6E-09 in 50 years. Thus, the reduction in ^{10}B concentration in the fixed neutron absorber by neutron absorption is negligible. In addition, the results presented in Subsection 3.4.4.3.1.8 demonstrate that the sheathing, which affixes the fixed neutron absorber panel, remains in place during all credible accident conditions, and thus, the fixed neutron absorber panel remains permanently fixed. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 1 of this FSAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel positioning was judged to be not significant. Starting in Revision 2 of this FSAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for all new or changed conditions. The calculations in this subsection serve to determine for which of these conditions the eccentric positioning of assemblies in the fuel storage locations results in a higher maximum k_{eff} value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum k_{eff} value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4. All other calculations throughout this chapter, such as studies to determine bounding fuel dimensions, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- Cell Center Configuration: All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;
- Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and
- Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be $(1/5)^{24}$ or approximately 10^{-17} for the MPC-24, $(1/5)^{32}$ or approximately 10^{-23} for the MPC-32, and $(1/5)^{68}$ or approximately 10^{-48} for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.6, results are presented for all conditions that were introduced in Revision 2 of this FSAR, namely results for the MPC-24E/~~EF~~ with intact and damaged fuel at 5 wt% ^{235}U, for the MPC-32 with soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U, and for the MPC-32 with intact and damaged fuel. The table shows the maximum k_{eff} value for centered and the two eccentric configurations for each condition, and the difference in k_{eff} between the centered and eccentric positioning. The results and conclusions are summarized as follows:

- In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.
- For the MPC-24E/~~EF~~, moving the assemblies and DFCs towards the center of the basket also results in a minor reduction. The cell centered configuration is therefore bounding for this condition and is used in the design basis calculations reported in Section 6.1 and Section 6.4.
- For the MPC-32 cases listed in Table 6.3.6, the maximum reactivity is shown for the basket center configuration. However, for some of the cases with intact and damaged fuel in the MPC-32, the cell centered configuration results in a higher maximum reactivity. Therefore, both the cell centered and basket centered configuration are analyzed for the MPC-32 design basis calculation, and the higher results are listed in the tables in Section 6.1. and 6.4. This applies to the cases with intact and damaged fuel, and to cases with intact fuel only and soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U.

Table 6.3.1

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter†	Δk for Maximum Tolerance		Action/Modeling Assumption
	MPC-24‡	MPC-68	
Reduce Fixed Neutron Absorber Width to Minimum	N/A††† min.= nom.= 7.5" and 6.25"	N/A††† min. = nom. = 4.75"	Assume minimum fixed neutron absorber width
Increase UO_2 Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO_2 density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min.= 8.86" nom. = 8.92"	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" nom. = 8.92"	-0.0030 max. = 6.113" nom. = 6.053"	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" nom. = 1.15"	N/A	Assume minimum water gap in the MPC-24

† Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

‡ Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

††† The fixed neutron absorber width for the MPC-68 is 4.75" +0.125", -0" , the fixed neutron absorber widths for the MPC-24 are 7.5" +0.125", -0" and 6.25" +0.125" –0" (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	Δk Maximum Tolerance		Action/Modeling Assumption
	MPC-24‡	MPC-68	
Increase in Temperature 20°C 40°C 70°C 100°C	 Ref. -0.0030 -0.0089 -0.0162	 Ref. -0.0039 -0.0136 -0.0193	Assume 20°C
10% Void in Moderator 20°C with no void 20°C 100°C	 Ref. -0.0251 -0.0412	 Ref. -0.0241 -0.0432	Assume no void
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

‡ Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated k_{eff}
MPC-24†† (17x17A01 @4.0% Enrichment)						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9325±0.0008†††
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9300±0.0008
nominal	(10.906")	nom. - 0.04"	(8.88")	nom. + 0.05"	(0.3625")	0.9305±0.0007
MPC-68 (8x8C04 @4.2% Enrichment)						
minimum	(6.43")	minimum	(5.993")	nominal	(1/4")	0.9307±0.0007
nominal	(6.49")	nominal	(6.053")	nominal	(1/4")	0.9274±0.0007
maximum	(6.55")	maximum	(6.113")	nominal	(1/4")	0.9272±0.0008
nom. + 0.05"	(6.54")	nominal	(6.053")	nom. + 0.05"	(0.30")	0.9267±0.0007

Notes:

1. Values in parentheses are the actual value used.

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

†† All calculations for the MPC-24 assume minimum water gap thickness (1.09").

††† Numbers are 1σ statistical uncertainties.

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-24 (17x17A @5.0% Enrichment) 400ppm soluble boron			
nominal (10.906")	maximum (8.98")	nominal (5/16")	0.9236±0.0007††
maximum (10.966")	maximum (8.98")	nominal (5/16")	0.9176±0.0008
minimum (10.846")	nominal (8.92")	nominal (5/16")	0.9227±0.0010
minimum (10.846")	minimum (8.86")	nominal (5/16")	0.9159±0.0008
nominal (10.906")	nominal-0.04" (8.88")	nom.+0.05" (0.3625")	0.9232±0.0009
nominal (10.906")	nominal (8.92")	nominal (5/16")	0.9158±0.0007
MPC-32 (17x17A @5.0% Enrichment) 2600 ppm soluble boron †††			
minimum (9.158")	minimum (8.69")	nominal (9/32")	0.9085±0.0007
nominal (9.218")	nominal (8.75")	nominal (9/32")	0.9028±0.0007
maximum (9.278")	maximum (8.81")	nominal (9/32")	0.8996±0.0008
nominal+0.05" (9.268")	nominal (8.75")	nominal+0.05" (0.331")	0.9023±0.0008
minimum+0.05"(9.208")	minimum (8.69")	nominal+0.05" (0.331")	0.9065±0.0007
maximum (9.278")	Maximum-0.05" (8.76")	nominal+0.05" (0.331")	0.9030±0.0008

Notes:

1. Values in parentheses are the actual value used.

† 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.

Table 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Pitch	Box I.D.	Box Wall Thickness	Water-Gap Flux Trap
MPC-24	nominal (10.906")	maximum (8.98")	nominal (5/16")	minimum (1.09")
MPC-24E	nominal (10.847")	maximum (8.81", 9.11" for DFC Positions)	nominal (5/16")	minimum (1.076", 0.776" for DFC Positions)
MPC-32	Minimum (9.158")	Minimum† (8.69")	Nominal (9/32")	N/A
MPC-68	minimum (6.43")	Minimum (5.993")	nominal (1/4")	N/A

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

Table 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

MPC-24, MPC-24E and MPC-32		
UO_2 5.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.696E-02	1.185E-01
92235	1.188E-03	4.408E-02
92238	2.229E-02	8.374E-01
UO_2 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
BORAL (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24E and MPC-32)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

METAMIC (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.648 (MPC-24)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	6.314E-03	3.965E-02
5011	2.542E-02	1.755E-01
6012	7.932E-02	5.975E-02
13027	4.286E-02	7.251E-01
METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646 (MPC-24E and MPC-32)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

BORATED WATER, 300 PPM, DENSITY (g/cc)=1.00		
Nuclide	**Atom-Density**	**Wt. Fraction**
5010	3.248E-06	5.400E-05
5011	1.346E-05	2.460E-04
1001	6.684E-02	1.1186E-01
8016	3.342E-02	8.8784E-01
BORATED WATER, 400PPM, DENSITY (g/cc)=1.00		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	4.330E-06	7.200E-05
5011	1.794E-05	3.280E-04
1001	6.683E-02	1.1185E-01
8016	3.341E-02	8.8775E-01
BORATED WATER, 1900PPM, DENSITY (g/cc)=1.00		
Nuclide	Atom-Density	**Wgt. Fraction**
5010	2.057E-05	3.420E-04
5011	8.522E-05	1.558E-03
1001	6.673E-02	1.1169E-01
8016	3.336E-02	8.8641E-01
BORATED WATER, 2600PPM, DENSITY (g/cc)=0.93		
Nuclide	Atom-Density	**Wgt. Fraction**
5010	2.618e-05	4.680E-04
5011	1.085e-04	2.132E-03
1001	6.201e-02	1.1161E-01
8016	3.101e-02	8.8579E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

MPC-68		
UO_2 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
UO_2 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
MOX FUEL†, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

† The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	6.541E-03	4.110E-02
5011	2.633E-02	1.819E-01
6012	8.217E-03	6.193E-02
13027	4.223E-02	7.151E-01
FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.798E-02	1.212E-01
92235	4.001E-04	1.484E-02
92238	2.742E-05	1.030E-03
90232	2.357E-02	8.630E-01
COMMON MATERIALS		
ZR CLAD, DENSITY (g/cc) = 6.550		
Nuclide	**Atom-Density**	**Wgt. Fraction**
40000	4.323E-02	1.000E+00
MODERATOR (H_2O), DENSITY (g/cc) = 1.000		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

STAINLESS STEEL, DENSITY (g/cc) = 7.840		
Nuclide	**Atom-Density**	**Wgt. Fraction**
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
ALUMINUM, DENSITY (g/cc) = 2.700		
Nuclide	**Atom-Density**	**Wgt. Fraction**
13027	6.026E-02	1.000E+00

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

CONCRETE, DENSITY (g/cc) = 2.35		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	8.806E-03	6.000E-03
8016	4.623E-02	5.000E-01
11000	1.094E-03	1.700E-02
13027	2.629E-04	4.800E-03
14000	1.659E-02	3.150E-01
19000	7.184E-04	1.900E-02
20000	3.063E-03	8.300E-02
26000	3.176E-04	1.200E-02
LEAD, DENSITY (g/cc) = 11.34		
Nuclide	**Atom-Density**	**Wgt. Fraction**
82000	3.296E-02	1.0
HOLTITE-A, DENSITY (g/cc) = 1.61		
1001	5.695E-02	5.920E-02
5010	1.365E-04	1.410E-03
5011	5.654E-04	6.420E-03
6012	2.233E-02	2.766E-01
7014	1.370E-03	1.980E-02
8016	2.568E-02	4.237E-01
13027	7.648E-03	2.129E-01

Table 6.3.5

REACTIVITY EFFECT OF SHEATHING THICKNESS FOR THE MPC-32

Assembly Class	**Enrichment (wt% ^{235}U)**	**Soluble Boron (ppm)**	**Maximum k_{eff}**		**Difference in Maximum k_{eff}**
			Sheathing 0.075" Min. Cell ID 8.69"	**Sheathing 0.035" Min. Cell ID 8.73"**	
15x15F	5.0	2600	0.9483	0.9476	-0.0008
15x15F	1.7	0	0.8914	0.8909	-0.0005

Table 6.3.6

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT (FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

CASE	Contents centered (Reference)	Content moved towards center of basket		Content moved towards basket periphery	
	Maximum k_{eff}	Maximum k_{eff}	k_{eff} Difference to Reference	Maximum k_{eff}	k_{eff} Difference to Reference
MPC-24E/~~EF~~, Intact Fuel and Damaged Fuel/Fuel Debris, 5% Enrichment, 600ppm Soluble Boron	0.9185	0.9178	-0.0007	0.9132	-0.0053
MPC-32/~~32F~~, Intact Fuel, Assembly Class 16x16A, 4.1% Enrichment, 14~~3~~00ppm Soluble Boron	0.9332~~429~~	0.9367~~468~~	0.0035~~39~~	0.8992~~9068~~	-0.0340~~61~~
MPC-32/~~32F~~, Intact Fuel, Assembly Class 15x15B, 5.0% Enrichment, 2400ppm Soluble Boron	0.9473	0.9493	0.0020	0.9306	-0.0167
MPC-32/~~32F~~, Intact Fuel and Damaged Fuel/Fuel Debris, Assembly Class 15x15F (Intact), 5% Enrichment, 2900ppm Soluble Boron	0.9378	0.9397	0.0019	0.9277	-0.0101

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational or Experimental Method

6.4.1.1 Basic Criticality Safety Calculations

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, a minimum of 5,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket in the HI-STORM 100 System.

CASMO-3 [6.1.9] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO-3 has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO-3 calculations.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Tables 6.1.1 through 6.1.8 . These calculations were based on the assumption that the HI-STORM 100 System (HI-TRAC transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

6.4.2.1 Internal and External Moderation

As required by NUREG-1536, calculations in this section demonstrate that the HI-STORM 100 System remains subcritical for all credible conditions of moderation.

6.4.2.1.1 Unborated Water

With a neutron absorber present (i.e., the fixed neutron absorber sheets or the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask were made to confirm that the phenomenon does not occur with low density water inside or outside the casks.

Calculations for the MPC designs with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case 1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Figure 6.4.10 plots calculated k_{eff} values ($\pm 2\sigma$) as a function of internal moderator density for both MPC designs with 100% external moderator density (i.e., full water reflection). Results listed in Table 6.4.1 support the following conclusions:

- For each type of MPC, the calculated k_{eff} for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and

- For each type of MPC, reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STORM 100 System.

For each of the MPC designs, the maximum k_{eff} values are shown to be less than or statistically equal to that of a single internally flooded unreflected cask and are below the regulatory limit of 0.95.

6.4.2.1.2 Borated Water

With the presence of a soluble neutron absorber in the water, the discussion in the previous section is not always applicable. Calculations were made to determine the optimum moderator density for the MPC designs that require a minimum soluble boron concentration.

Calculations for the MPC designs with various internal moderator densities are shown in Table 6.4.6. As shown in the previous section, the external moderator density has a negligible effect on the reactivity, and is therefore not varied. Water containing soluble boron has a slightly higher density than pure water. Therefore, water densities up to 1.005 g/cm^3 were analyzed for the higher soluble boron concentrations. Additionally, for the higher soluble boron concentrations, analysis have been performed with empty (voided) guide tubes. This variation is discussed in detail in Section 6.4.8. Results listed in the Table 6.4.6 support the following conclusions:

- For all cases with a soluble boron concentration of up to 1900ppm, and for a soluble boron concentration of 2600ppm assuming voided guide tubes, the conclusion of the Section 6.4.2.1.1 applies, i.e. the maximum reactivity is corresponds to 100% moderator density.

- For 2600ppm soluble boron concentration with filled guide tubes, the results presented in Table 6.4.6 indicate that there is a maximum of the reactivity somewhere between 0.90 g/cm^3 and 1.00 g/cm^3 moderator density. However, a distinct maximum can not be identified, as the reactivities in this range are very close. For the purpose of the calculations with 2600ppm soluble boron concentration, a moderator density of 0.93 g/cm^3 was chosen, which corresponds to the highest calculated reactivity listed in Table 6.4.6.

The calculations documented in this chapter also use soluble boron concentrations other than 1900 ppm and 2600 ppm in the MPC-32/~~32F~~. For the MPC-32 loaded with intact fuel only, soluble boron concentrations between 1300 ppm and 2600 ppm are used. For the MPC-32/~~32F~~ loaded with intact fuel, damaged fuel and fuel debris, soluble boron concentrations between 1500 ppm and 2900 ppm are used. In order to determine the optimum moderation condition for each assembly class at the corresponding soluble boron level, evaluations are performed with filled and voided guide tubes, and for water densities of 1.0 g/cm^3 and 0.93 g/cm^3 for each class and enrichment level. Results for the MPC-32 loaded with intact fuel only are listed in Table 6.4.10 for an initial enrichment of 5.0 wt% ^{235}U and in Table 6.4.11 for an initial enrichment of 4.1 wt% ^{235}U. Corresponding results for the MPC-32~~/32F~~ loaded with intact fuel, damaged fuel and fuel debris are listed in Table 6.4.14. The highest value listed in these tables for each assembly class is listed as the bounding value in Section 6.1.

6.4.2.2 Partial Flooding

As required by NUREG-1536, calculations in this section address partial flooding in the HI-STORM 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

6.4.2.3 Clad Gap Flooding

As required by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e. different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 11). Because the fuel cladding temperatures remain below their design limits (as demonstrated in Chapter 4) and the inertial loading remains below 63g's (the inertial loadings associated with the design basis drop accidents discussed in Chapter 11 are limited to 45g's), the cladding remains intact (see Section 3.5). For damaged fuel assemblies and fuel debris, the assemblies or debris are pre-loaded into stainless steel Damaged Fuel Containers fitted with 250x250 fine mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. This condition would be the result of the water tension across the mesh screens. The maximum water level inside the DFCs for this condition is calculated from the dimensions of the mesh screen and the surface tension of water. The wetted perimeter of the screen openings is 50 ft per square inch of screen. With a surface tension of water of 0.005 lbf/ft, this results in a maximum pressure across the screen of 0.25 psi, corresponding to a maximum water height in the DFC of 7 inches. For added conservativism, a value of 12 inches is used. Assuming this condition, calculations are performed for all three possible DFC configurations:

- MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)
- MPC-68 ~~or MPC-68FF~~ with 16 DFCs (All BWR Assembly Classes)
- MPC-24E ~~or MPC-24EF~~ with 4 DFCs (All PWR Assembly Classes)
- MPC-32 ~~or MPC-32F~~ with 8 DFCs (All PWR Assembly Classes)

For each configuration, the case resulting in the highest maximum k_{eff} for the fully flooded condition (see Section 6.4.4) is re-analyzed assuming the preferential flooding condition. For these analyses, the lower 12 inches of the active fuel in the DFCs and the water region below the active fuel (see Figure 6.3.7) are filled with full density water (1.0 g/cc). The remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). Table 6.4.4 lists the maximum k_{eff} for the four configurations in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{eff} than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC confinement boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Section 6.4.4.

6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 11 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC transfer cask and minor damage to the concrete radiation shield for the HI-STORM storage cask, which have no adverse effect on the design parameters important to criticality safety.

As reported in Chapter 3, Table 3.4.4, the minimum factor of safety for either MPC as a result of the hypothetical cask drop or tip-over accident is 1.1 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change will be insignificant compared to the characteristic dimension of the flux trap.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM 100 System is in full compliance with the requirement of 10CRF72.124, which states that "before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety."

6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) are presented in section 6.2 and summarized in Section 6.1. To demonstrate the applicability of the HI-STAR analyses, results of the design basis criticality safety calculations for the HI-STAR cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-3) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff}^{max} = k_c + K_c\sigma_c + Bias + \sigma_B$$

where:

⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;

⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final k_{eff} value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;

⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code (MCNP4a or KENO5a);

⇒ *Bias* is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and

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

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

6.4.4 Damaged Fuel and Fuel Debris

Damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. Five (5) different DFC types with different cross sections are analyzed. Three (3) of these DFCs are designed for BWR fuel assemblies, two (2) are designed for PWR fuel assemblies. Two of the DFCs for BWR fuel are specifically designed for fuel assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. These assemblies have a smaller cross section, a shorter active length and a low initial enrichment of 2.7 wt% ^{235}U, and therefore a low reactivity. The analysis for these assembly classes is presented in the following Section 6.4.4.1. The remaining three DFCs are generic DFCs designed for all BWR and PWR assembly classes. The criticality analysis for these generic DFCs is presented in Section 6.4.4.2.

6.4.4.1 MPC-68 *or*~~,~~ MPC-68F ~~or MPC-68FF~~ loaded with Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

This section only addresses criticality calculations and results for assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A, loaded into the MPC-68~~, MPC-68F~~ or MPC-68F~~F~~. Up to 68 DFCs with these assembly classes are permissible to be loaded into the MPC. Two different DFC types with slightly different cross-sections are analyzed. ~~DFCs containing fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing damaged fuel assemblies may be stored in either the MPC-68, MPC-68F or MPC-68FF.~~ Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 ~~and MPC-68FF~~ ha~~ve~~s a higher specified ^{10}B loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68 ~~or MPC-68FF~~. Although the maximum planar-average enrichment of the damaged fuel is limited to

2.7% ^{235}U as specified in Section 2.1.9, analyses have been made for three possible scenarios, conservatively assuming fuel[††] of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.2 through 6.4.8.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This configuration is illustrated in Figure 6.4.9.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.5, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in ^{238}U neutron capture (higher effective resonance integral for ^{238}U absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.5.

The analyses performed and summarized in Table 6.4.5 provides the relative magnitude of the effects on the reactivity. This information coupled with the maximum k_{eff} values listed in Table 6.1.3 and the conservatism in the analyses, demonstrate that the maximum k_{eff} of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of $k_{eff} < 0.95$.

6.4.4.2 Generic BWR and PWR Damaged Fuel and Fuel Debris

The MPC-24E, ~~MPC-24EF,~~ MPC-32~~,~~ ~~MPC-32F, MPC-68~~ and MPC-68~~FF~~ are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into generic DFCs. The number of generic DFCs is limited to 16 for the MPC-68~~and MPC-68FF~~, to 4 for the MPC-24E~~and MPC-24EF~~, and to 8 for the MPC-32~~and MPC-32F~~. The permissible locations of the DFCs are shown in Figure 6.4.11 for the MPC-68~~/68FF~~, in Figure 6.4.12 for the MPC-24E~~/24EF~~ and in Figure 6.4.16 for the MPC-32~~/32F~~.

[††] 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

Damaged fuel assemblies are assemblies with known or suspected cladding defects greater than pinholes or hairlines, or with missing rods, but excluding fuel assemblies with gross defects (for a full definition see Table 1.0.1). Therefore, apart from possible missing fuel rods, damaged fuel assemblies have the same geometric configuration as intact fuel assemblies and consequently the same reactivity. Missing fuel rods can result in a slight increase of reactivity. After a drop accident, however, it can not be assumed that the initial geometric integrity is still maintained. For a drop on either the top or bottom of the cask, the damaged fuel assemblies could collapse. This would result in a configuration with a reduced length, but increased amount of fuel per unit length. For a side drop, fuel rods could be compacted to one side of the DFC. In either case, a significant relocation of fuel within the DFC is possible, which creates a greater amount of fuel in some areas of the DFC, whereas the amount of fuel in other areas is reduced. Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

In the cases of fuel debris or relocated damaged fuel, there is the potential that fuel could be present in axial sections of the DFCs that are outside the basket height covered with the fixed neutron absorber. However, in these sections, the DFCs are not surrounded by any intact fuel, only by basket cell walls, non-fuel hardware, water and for the MPC-68/68FF by a maximum of one other DFC. Studies have shown that this condition does not result in any significant effect on reactivity, compared to a condition where the damaged fuel and fuel debris is restricted to the axial section of the basket covered by the fixed neutron absorber. All calculations for generic BWR and PWR damaged fuel and fuel debris are therefore performed assuming that fuel is present only in the axial sections covered by the fixed neutron absorber, and the results are directly applicable to any situation where damaged fuel and fuel debris is located outside these sections in the DFCs.

To address all the situations listed above and identify the configuration or configurations leading to the highest reactivity, it is impractical to analyze a large number of different geometrical configurations for each of the fuel classes. Instead, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following sections.

All calculations for generic damaged fuel and fuel debris are performed using a full cask model with the maximum permissible number of Damaged Fuel Containers. For the MPC-68~~ and MPC-68FF~~, the model therefore contains 52 intact assemblies, and 16 DFCs in the locations shown in Figure 6.4.11. For the MPC-24E~~ and MPC-24EF~~, the model consists of 20 intact assemblies, and 4 DFCs in the locations shown in Figure 6.4.12. For the MPC-32~~ and MPC-32~~, the model consists of 24 intact assemblies, and 8 DFCs in the locations shown in Figure 6.4.16. The bounding assumptions regarding the intact assemblies and the modeling of the damaged fuel and fuel debris in the DFCs are discussed in the following sections.

Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term ' damaged fuel' as used throughout this chapter designates both damaged fuel and fuel debris.

6.4.4.2.1 Bounding Intact Assemblies

Intact BWR assemblies stored together with DFCs are limited to a maximum planar average enrichment of 3.7 wt% ^{235}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 shows results for enrichments of 4.1 wt% and 5.0 wt%, respectively, for all fuel classes. Since different minimum soluble boron concentrations are used for different groups of assembly classes, the assembly class with the highest reactivity in each group is used as the intact assembly for the calculations with DFCs in the MPC-32. These assembly classes are

- 14x14C for all 14x14 assembly classes;
- 15x15B for assembly classes 15x15A, B, C and G;
- 15x15F for assembly classes 15x15D, E, F and H;
- 16x16A; and
- 17x17C for all 17x17 assembly classes.

6.4.4.2.2 Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e. all cladding and other structural material in the DFC is replaced by water.

- For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.

- The active length of these rods is chosen to be the maximum active fuel length of all fuel assemblies listed in Section 6.2, which is 155 inch for BWR fuel and 150 inch for PWR fuel.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 9 (3x3) and 189 (17x17) for BWR fuel, and between 64 (8x8) and 729 (27x27) for PWR fuel.

- Analyses are performed for the minimum, maximum and typical pellet diameter of PWR and BWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

This is also a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 155 inch (BWR) or 150 inch (PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

To demonstrate the level of conservatism, additional analyses are performed with the DFC containing various realistic assembly configurations such as intact assemblies, assemblies with missing fuel rods and collapsed assemblies, i.e. assemblies with increased number of rods and decreased rod pitch.

As discussed in Section 6.4.4.2, all calculations are performed for full cask models, containing the maximum permissible number of DFCs together with intact assemblies.

As an example of the damaged fuel model used in the analyses, Figure 6.4.17 shows the basket cell of an MPC-32 with a DFC containing a 17x17 array of bare fuel rods.

Graphical presentations of the calculated maximum k_{eff} for typical cases as a function of the fuel mass per unit length of the DFC are shown in Figures 6.4.13 (BWR) and 6.4.14 (PWR, MPC-24E/EF with pure water). The results for the bare fuel rods show a distinct peak in the maximum k_{eff} at about 2 kg UO_2/inch for BWR fuel, and at about 3.5 kgUO_2/inch for PWR fuel.

The realistic assembly configurations are typically about 0.01 (delta-k) or more below the peak results for the bare fuel rods, demonstrating the conservatism of this approach to model damaged fuel and fuel debris configurations such as severely damaged assemblies and bundles of fuel rods.

For fuel debris configurations consisting of bare fuel pellets only, the fuel mass per unit length would be beyond the value corresponding to the peak reactivity. For example, for DFCs filled with a mixture of 60 vol% fuel and 40 vol% water the fuel mass per unit length is 3.36 kgUO_2/inch for the BWR DFC and 7.92 kgUO_2/inch for the PWR DFC. The corresponding reactivities are significantly below the peak reactivies. The difference is about 0.005 (delta-k) for BWR fuel and 0.01 (delta-k) or more for PWR fuel. Furthermore, the filling height of the DFC would be less than 70 inches in these examples due to the limitation of the fuel mass per basket position, whereas the calculation is conservatively performed for a height of 155 inch (BWR) or 150 inch (PWR). These results demonstrate that even for the fuel debris configuration of bare fuel pellets, the model using bare fuel rods is a conservative approach.

6.4.4.2.3 Distributed Enrichment in BWR Fuel

BWR fuel usually has an enrichment distribution in each planar cross section, and is characterized by the maximum planar average enrichment. For intact fuel it has been shown that using the average enrichment for each fuel rod in a cross section is conservative, i.e. the reactivity is higher than calculated for the actual enrichment distribution (See Appendix 6.B). For damaged fuel assemblies, additional configurations are analyzed to demonstrate that the distributed enrichment does not have a significant impact on the reactivity of the damaged assembly under accident conditions. Specifically, the following two scenarios were analyzed:

- As a result of an accident, fuel rods with lower enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the top part, but at the same time the amount of fuel in that area is reduced compared to the intact assembly.
- As a result of an accident, fuel rods with higher enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the bottom part, and at the same time the amount of fuel in that area is increased compared to the intact assembly, leading to a reduction of the water content.

In both scenarios, a compensation of effects on reactivity is possible, as the increase of reactivity due to the increased planar average enrichment might be offset by the possible reduction of reactivity due to the change in the fuel to water ratio. A selected number of calculations have been performed for these scenarios and the results show that there is only a minor change in reactivity. These calculations are shown in Figure 6.4.13 in the group of the explicit assemblies. Consequently, it is appropriate to qualify damaged BWR fuel assemblies and fuel debris based on the maximum planar average enrichment. For assemblies with missing fuel rods, this maximum planar average enrichment has to be determined based on the enrichment and number of rods still present in the assembly when loaded into the DFC.

6.4.4.2.4 Results for *the* MPC-68 ~~and MPC-68FF~~

The MPC-68 ~~and MPC-68FF~~ allows the storage of up to sixteen DFCs in the shaded cells on the periphery of the basket shown in Figure 6.4.11. ~~In the MPC-68FF~~*Additionally*, up to 8 of these cells may contain DFCs with fuel debris. The various configurations outlined in Sections 6.4.4.2.2 and 6.4.4.2.3 are analyzed with an enrichment of the intact fuel of 3.7% ^{235}U and an enrichment of damaged fuel or fuel debris of 4.0% ^{235}U. For the intact assembly, the bounding assembly of the 9x9E and 9x9F fuel classes was chosen. This assembly has the highest reactivity of all BWR assembly classes for the initial enrichment of 3.7 wt% ^{235}U, as demonstrated in Table 6.4.7. The results for the various configurations are summarized in Figure 6.4.13 and in Table 6.4.8. Figure 6.4.13 shows the maximum k_{eff}, including bias and calculational uncertainties, for various actual and hypothetical damaged fuel or fuel debris configurations as a function of the fuel mass per unit length of the DFC. Table 6.4.8 lists the highest maximum k_{eff} for the various configurations. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.4.4.2.5 Results for *the* MPC-24E ~~and MPC-24EF~~

The MPC-24E allows the storage of up to four DFCs with damaged fuel *or fuel debris* in the four outer fuel baskets cells shaded in Figure 6.4.12. ~~The MPC-24EF allows storage of up to four DFCs with damaged fuel or fuel debris in these locations.~~ These locations are designed with a larger box ID to accommodate the DFCs. For an enrichment of 4.0 wt% ^{235}U for the intact fuel, damaged fuel and fuel debris, and assuming no soluble boron, the results for the various configurations outlined in Section 6.4.4.2.2 are summarized in Figure 6.4.14 and in Table 6.4.9. Figure 6.4.14 shows the maximum k_{eff}, including bias and calculational uncertainties, for various actual and hypothetical damaged fuel and fuel debris configurations as a function of the fuel mass per unit length of the DFC. For the intact assemblies, the 15x15H assembly class was chosen. This assembly class has the highest reactivity of all PWR assembly classes for a given initial enrichment. This is demonstrated in Table 6.1.4. Table 6.4.9 lists the highest maximum k_{eff} for the various configurations. All maximum k_{eff} values are below the 0.95 regulatory limit.

For an enrichment of 5.0 wt% ^{235}U for the intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration of 600 ppm is required. For this condition, calculations are performed for various hypothetical fuel debris configurations (i.e. bare fuel rods) as a function of the fuel mass per unit length of the DFC. Additionally, calculations are performed with reduced water densities in the DFC. The various conditions of damaged fuel, such as assemblies with missing rods or collapsed assemblies, were not analyzed, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods. Again, the 15x15H assembly class was chosen as the intact assembly since this assembly class has the highest reactivity of all PWR assembly classes as demonstrated in Table 6.1.4. The results are summarized in Table 6.4.12. Similar to the calculations with pure water (see Figure 6.4.14), the results for borated water show a distinct peak of the maximum k_{eff} as a function of the fuel mass per unit length. Therefore, for each condition, the table lists only the highest maximum k_{eff}, including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the reactivity decreases with decreasing water density. This demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.4.4.2.6 Results for *the* MPC-32 ~~and MPC-32F~~

The MPC-32 allows the storage of up to eight DFCs with damaged fuel *or fuel debris* in the outer fuel basket cells shaded in Figure 6.4.16. ~~The MPC-32F allows storage of up to eight DFCs with damaged fuel or fuel debris in these locations.~~ For the MPC-32~~-and MPC-32F~~, additional cases are analyzed due to the high soluble boron level required for this basket:

- The assembly classes of the intact assemblies are grouped, and minimum required soluble boron levels are determined separately for each group. The analyses are performed for the bounding assembly class in each group. The bounding assembly classes are listed in Section 6.4.4.2.1.

- Evaluations of conditions with voided and filled guide tubes and various water densities in the MPC and DFC are performed to identify the most reactive condition.

In general, all calculations performed for the MPC-32 show the same principal behavior as for the MPC-24 (see Figure 6.4.14), i.e. the reactivity as a function of the fuel mass per unit length for the bare fuel rod array shows a distinct peak. Therefore, for each condition analyzed, only the highest maximum k_{eff}, i.e. the calculated peak reactivity, is listed in the tables. Evaluations of different diameters of the bare fuel pellets and the reduced water density in the DFC have been performed for a representative case using the 15x15F assembly class as the intact assembly, with voided guide tubes, a water density of 1.0 g/cc in the DFC and MPC, -2900 ppm soluble boron, and an enrichment of 5.0 wt% ^{235}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.

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 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 $k_{eff} < 0.95$.

6.4.7 Sealed Rods replacing BWR Water Rods

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

6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

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

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes are filled or assumed empty. For the MPC-24 with 400 ppm and the MPC-32 with 1900 ppm, voiding the guide tubes results in a reduction of reactivity. All calculations for the MPC-24 and MPC-24E are therefore performed with water in the guide tubes. For the MPC-32 with 2600 ppm, the reactivity for voided guide tubes slightly exceeds the reactivity for filled guide tubes. However, this effect is not consistent across all assembly classes. Table 6.4.10, Table 6.4.11 and Table 6.4.14 show results with filled and voided guide tubes for all assembly classes in the MPC-32/32F at 4.1 wt% ^{235}U and 5.0 wt% ^{235}U. Some classes show an increase, other classes show a decrease as a result of voiding the guide tubes. Therefore, for the results presented in the Section 6.1, Table 6.1.5, Table 6.1.6 and Table 6.1.12, the maximum value for each class is chosen for each enrichment level.

In summary, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

6.4.9 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a keff less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

6.4.10 Applicability of HI-STAR Analyses to HI-STORM 100 System

Calculations previously supplied to the NRC in applications for the HI-STAR 100 System (Docket Numbers 71-9261 and 72-1008) are directly applicable to the HI-STORM storage and HI-TRAC transfer casks. The MPC designs are identical. The cask systems differ only in the overpack shield material. The limiting condition for the HI-STORM 100 System is the fully flooded HI-TRAC transfer cask. As demonstrated by the comparative calculations presented in Tables 6.1.1 through 6.1.8, the shield material in the overpack (steel and lead for HI-TRAC, steel for HI-STAR) has a negligible impact on the eigenvalue of the cask systems. As a result, this analysis for the 125-ton HI-TRAC transfer cask is applicable to the 100-ton HI-TRAC transfer cask. In all cases, for the reference fuel assemblies, the maximum k_{eff} values are in good agreement and are conservatively less than the limiting k_{eff} value (0.95).

6.4.11 Fixed Neutron Absorber Material

The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorber materials: Boral or Metamic. Both materials are made of aluminum and B_4C powder. Boral has an inner core consisting of B_4C and aluminum between two outer layers consisting of aluminum only. This configuration is explicitly modeled in the criticality evaluation and shown in Figures 6.3.1 through 6.3.3 for each basket. Metamic is a single layer material with the same overall thickness and the same credited ^{10}B loading (in g/cm^2) for each basket. The majority of the criticality evaluations documented in this chapter are performed using Boral as the fixed neutron absorber. For a selected number of bounding cases, analyses are also performed using Metamic instead of Boral. The results for these cases are listed in Table 6.4.15, together with the corresponding result using Boral and the difference between the two materials for each case. Individual cases show small differences for the two materials. However, the differences are mostly below two times the standard deviation (the standard deviation is about 0.0008 for all cases in Table 6.4.15), indicating that the results are statistically equivalent. Furthermore, the average difference is well below one standard deviation, and all cases are below the regulatory limit of 0.95. In some cases listed in Table 6.4.15, the reactivity difference between Metamic and Boral might be larger than expected for two equivalent materials. Also, for four out of the five cases with MPC-24 type baskets, Metamic shows the higher reactivity, which could potentially indicate a trend rather than a statistical variation. Therefore, in order to confirm that the materials are equivalent, a second set of calculations was performed for Metamic, which was statistically independent from the set shown in Table 6.4.15. This was achieved by selecting a different starting value for the random number generator in the Monte Carlo calculations. The second set also shows some individual variations of the differences, and a low average difference. However, there is no apparent trend regarding the MPC-24 type baskets compared to the MPC-32 and MPC-68, and the maximum positive reactivity difference for Metamic in an MPC-24 type basket is only 0.0005. Overall, the calculations demonstrate that the two fixed neutron absorber materials are identical from a criticality perspective. All results obtained for Boral are therefore directly applicable to Metamic and no further evaluations using Metamic are required.

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS†

	Water Density		MCNP4a Maximum k_{eff} ††	
Case Number	Internal	External	MPC-24 (17x17A01 @ 4.0%)	MPC-68 (8x8C04 @ 4.2%)
1	100%	single cask	0.9368	0.9348
2	100%	100%	0.9354	0.9339
3	100%	70%	0.9362	0.9339
4	100%	50%	0.9352	0.9347
5	100%	20%	0.9372	0.9338
6	100%	10%	0.9380	0.9336
7	100%	5%	0.9351	0.9333
8	100%	0%	0.9342	0.9338
9	70%	0%	0.8337	0.8488
10	50%	0%	0.7426	0.7631
11	20%	0%	0.5606	0.5797
12	10%	0%	0.4834	0.5139
13	5%	0%	0.4432	0.4763
14	10%	100%	0.4793	0.4946

† For an infinite square array of casks with 60cm spacing between cask surfaces.

†† Maximum k_{eff} includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

MPC-24 (17x17A01 @ 4.0% ENRICHMENT) (no soluble boron)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
MPC-68 (8x8C04 @ 4.2% ENRICHMENT)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348
MPC-32 (15x15F @ 5.0 % ENRICHMENT) 2600ppm Soluble Boron			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.8927	31.25	0.9213
50	0.9215	50	0.9388
75	0.9350	68.75	0.9401
100	0.9445	100	0.9445

Notes:

1. All values are maximum k_{eff} which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded with unborated water	0.9368	0.9348

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.4

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Preferential Flooding	Fully Flooded
MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)	0.6560	0.7857
MPC-68 ~~or MPC-68FF~~ with 16 DFCs (All BWR Assembly Classes)	0.6646	0.9328
MPC-24E ~~or MPC-24EF~~ with 4 DFCs (All PWR Assembly Classes)	0.7895	0.9480
MPC-32 ~~or MPC-32~~ with 8 DFCs (All PWR Assembly Classes)	0.7213	0.9378

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.5

MAXIMUM k_{eff} VALUES† IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Maximum†† k_{eff}	
	DFC Dimensions: ID 4.93" THK. 0.12"	**DFC Dimensions: ID 4.81" THK. 0.11"**
6x6 Fuel Assembly		
6x6 Intact Fuel	0.7086	0.7016
w/32 Rods Standing	0.7183	0.7117
w/28 Rods Standing	0.7315	0.7241
w/24 Rods Standing	0.7086	0.7010
w/18 Rods Standing	0.6524	0.6453
Collapsed to 8x8 array	0.7845	0.7857
Dispersed Powder	0.7628	0.7440
7x7 Fuel Assembly		
7x7 Intact Fuel	0.7463	0.7393
w/41 Rods Standing	0.7529	0.7481
w/36 Rods Standing	0.7487	0.7444
w/25 Rods Standing	0.6718	0.6644

† These calculations were performed with a planar-average enrichment of 3.0% and a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative

†† Maximum k_{eff} includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.6

MAXIMUM k_{eff} VALUES WITH REDUCED BORATED WATER DENSITIES

Internal Water Density† in g/cm³	**Maximum k_{eff}**				
	MPC-24 (400ppm) @ 5.0 %	**MPC-32 (1900ppm) @ 4.1 %**		**MPC-32 (2600ppm) @ 5.0 %**	
Guide Tubes	filled	filled	void	filled	void
1.005	NC††	0.9403	0.9395	NC	0.9481
1.00	0.9314	0.9411	0.9400	0.9445	0.9483
0.99	NC	0.9393	0.9396	0.9438	0.9462
0.98	0.9245	0.9403	0.9376	0.9447	0.9465
0.97	NC	0.9397	0.9391	0.9453	0.9476
0.96	NC	NC	NC	0.9446	0.9466
0.95	0.9186	0.9380	0.9384	0.9451	0.9468
0.94	NC	NC	NC	0.9445	0.9467
0.93	0.9130	0.9392	0.9352	0.9465	0.9460
0.92	NC	NC	NC	0.9458	0.9450
0.91	NC	NC	NC	0.9447	0.9452
0.90	0.9061	0.9384	NC	0.9449	0.9454
0.80	0.8774	0.9322	NC	0.9431	0.9390
0.70	0.8457	0.9190	NC	0.9339	0.9259
0.60	0.8095	0.8990	NC	0.9194	0.9058
0.40	0.7225	0.8280	NC	0.8575	0.8410
0.20	0.6131	0.7002	NC	0.7421	0.7271
0.10	0.5486	0.6178	NC	0.6662	0.6584

† External moderator is modeled at 0%. This is consistent with the results demonstrated in Table 6.4.1.
†† NC: Not Calculated

Table 6.4.7

MAXIMUM k_{eff} VALUES FOR INTACT BWR FUEL ASSEMBLIES WITH A MAXIMUM PLANAR AVERAGE ENRICHMENT OF 3.7 wt% ^{235}U

Fuel Assembly Class	Maximum k_{eff}
6x6A	0.8287
6x6C	0.8436
7x7A	0.8399
7x7B	0.9109
8x8A	0.8102
8x8B	0.9131
8x8C	0.9115
8x8D	0.9125
8x8E	0.9049
8x8F	0.9233
9x9A	0.9111
9x9B	0.9134
9x9C	0.9103
9x9D	0.9096
9x9E	0.9237
9x9F	0.9237
9x9G	0.9005
10x10A	0.9158
10x10B	0.9156
10x10C	0.9152
10x10D	0.9182
10x10E	0.8970

Table 6.4.8

MAXIMUM k_{eff} VALUES IN THE GENERIC BWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt% ^{235}U FOR DAMAGED FUEL AND 3.7 wt% ^{235}U FOR INTACT FUEL

Model Configuration inside the DFC	**Maximum k_{eff}**
Intact Assemblies (4 assemblies analyzed)	0.9241
Assemblies with missing rods (7 configurations analyzed)	0.9240
Assemblies with distributed enrichment (4 configurations analyzed)	0.9245
Collapsed Assemblies (6 configurations analyzed)	0.9258
Regular Arrays of Bare Fuel Rods (31 configurations analyzed)	0.9328

Table 6.4.9

MAXIMUM k_{eff} VALUES IN THE MPC-24E/~~EF~~ WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt% ^{235}U AND NO SOLUBLE BORON.

Model Configuration inside the DFC	**Maximum k_{eff}**
Intact Assemblies (2 assemblies analyzed)	0.9340
Assemblies with missing rods (4 configurations analyzed)	0.9350
Collapsed Assemblies (6 configurations analyzed)	0.9360
Regular Arrays of Bare Fuel Rods (36 configurations analyzed)	0.9480

Table 6.4.10

MAXIMUM k_{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
16X16A	20~~19~~00	0.9377~~460~~	0.9375~~450~~	0.9429~~74~~	0.9389~~434~~
17x17A	2600	0.9105	0.9145	0.9160	0.9161
17x17B	2600	0.9345	0.9358	0.9371	0.9356
17X17C	2600	0.9417	0.9431	0.9437	0.9430

Table 6.4.11

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

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 4.1 %			
		Guide Tubes Filled		Guide Tubes Voided	
		1.0 g/cm³	0.93 g/cm³	1.0 g/cm³	0.93 g/cm³
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
14x14E	1300	0.7340	0.7204	n/a	n/a
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
16X16A	1*4*~~3~~00	0.9*367*~~468~~	0.9*347*~~425~~	0.9*375*~~433~~	0.93*08*~~84~~
17x17A	1900	0.9105	0.9111	0.9106	0.9091
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

Table 6.4.12

MAXIMUM k_{eff} VALUES IN THE MPC-24E/~~24EF~~ WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt% ^{235}U AND 600 PPM SOLUBLE BORON.

Water Density inside the DFC	**Bare Fuel Pellet Diameter**	**Maximum k_{eff}**
1.00	minimum	0.9185
1.00	typical	0.9181
1.00	maximum	0.9171
0.95	typical	0.9145
0.90	typical	0.9125
0.60	typical	0.9063
0.10	typical	0.9025
0.02	typical	0.9025

Table 6.4.13

MAXIMUM k_{eff} VALUES IN THE MPC-32/~~32F~~ WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt% ^{235}U, 2900 PPM SOLUBLE BORON AND THE 15x15F ASSEMBLY CLASS AS INTACT ASSEMBLY.

Water Density inside the DFC	Bare Fuel Pellet Diameter	Maximum k_{eff}
1.00	minimum	0.9374
1.00	typical	0.9372
1.00	maximum	0.9373
0.95	typical	0.9369
0.90	typical	0.9365
0.60	typical	0.9308
0.10	typical	0.9295
0.02	typical	0.9283

Table 6.4.14

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

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	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	4.1	1500	0.9330~~22~~	0.9332~~21~~	0.9348~~35~~	0.9333~~02~~
	5.0	2300	0.9212~~198~~	0.9246~~39~~	0.9283~~89~~	0.9299~~67~~
17x17A, B, C	4.1	2100	0.9284	0.9290	0.9294	0.9285
	5.0	2900	0.9308	0.9338	0.9355	0.9367

Table 6.4.15

COMPARISON OF MAXIMUM k_{eff} VALUES FOR DIFFERENT FIXED NEUTRON ABSORBER MATERIALS

Case	Maximum k_{eff}		Reactivity Difference
	BORAL	**METAMIC**	
MPC-68, Intact Assemblies	0.9457	0.9452	-0.0005
MPC-68, with 16 DFCs	0.9328	0.9315	-0.0013
MPC-68F with 68 DFCs	0.8021	0.8019	-0.0002
MPC-24, 0ppm	0.9478	0.9491	+0.0013
MPC-24, 400ppm	0.9447	0.9457	+0.0010
MPC-24E, Intact Assemblies, 0ppm	0.9468	0.9494	+0.0026
MPC-24E, Intact Assemblies, 300ppm	0.9399	0.9410	+0.0011
MPC-24E, with 4 DFCs, 0ppm	0.9480	0.9471	-0.0009
MPC-32, Intact Assemblies, 1900ppm	0.9411	0.9397	-0.0014
MPC-32, Intact Assemblies, 2600ppm	0.9483	0.9471	-0.0012
Average Difference			**+0.0001**

APPENDIX 6.C: CALCULATIONAL SUMMARY

The following table lists the maximum k_{eff} (including bias, uncertainties, and calculational statistics), MCNP calculated k_{eff}, standard deviation, and energy of average lethargy causing fission (EALF) for each of the candidate fuel types and basket configurations.

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

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A01	HI-STAR	0.9295	0.9252	0.0008	0.2084
14x14A02	HI-STAR	0.9286	0.9242	0.0008	0.2096
14x14A03	HI-STORM	0.3080	0.3047	0.0003	3.37E+04
14x14A03	HI-TRAC	0.9283	0.9239	0.0008	0.2096
14x14A03	HI-STAR	0.9296	0.9253	0.0008	0.2093
14x14B01	HI-STAR	0.9159	0.9117	0.0007	0.2727
14x14B02	HI-STAR	0.9169	0.9126	0.0008	0.2345
14x14B03	HI-STAR	0.9110	0.9065	0.0009	0.2545
14x14B04	HI-STAR	0.9084	0.9039	0.0009	0.2563
B14x14B01	HI-TRAC	0.9237	0.9193	0.0008	0.2669
B14x14B01	HI-STAR	0.9228	0.9185	0.0008	0.2675
14x14C01	HI-TRAC	0.9273	0.9230	0.0008	0.2758
14x14C01	HI-STAR	0.9258	0.9215	0.0008	0.2729
14x14C02	HI-STAR	0.9265	0.9222	0.0008	0.2765
14x14C03	HI-TRAC	0.9274	0.9231	0.0008	0.2839
14x14C03	HI-STAR	0.9287	0.9242	0.0009	0.2825

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

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14D01	HI-TRAC	0.8531	0.8488	0.0008	0.3316
14x14D01	HI-STAR	0.8507	0.8464	0.0008	0.3308
14x14E01	HI-STAR	0.7598	0.7555	0.0008	0.3890
14x14E02	HI-TRAC	0.7627	0.7586	0.0007	0.3591
14x14E02	HI-STAR	0.7627	0.7586	0.0007	0.3607
14x14E03	HI-STAR	0.6952	0.6909	0.0008	0.2905
15x15A01	HI-TRAC	0.9205	0.9162	0.0008	0.2595
15x15A01	HI-STAR	0.9204	0.9159	0.0009	0.2608
15x15B01	HI-STAR	0.9369	0.9326	0.0008	0.2632
15C15B02	HI-STAR	0.9338	0.9295	0.0008	0.2640
15x15B03	HI-STAR	0.9362	0.9318	0.0008	0.2632
15x15B04	HI-STAR	0.9370	0.9327	0.0008	0.2612
15x15B05	HI-STAR	0.9356	0.9313	0.0008	0.2606
15x15B06	HI-STAR	0.9366	0.9324	0.0007	0.2638
B15x15B01	HI-TRAC	0.9387	0.9344	0.0008	0.2616
B15x15B01	HI-STAR	0.9388	0.9343	0.0009	0.2626
15x15C01	HI-STAR	0.9255	0.9213	0.0007	0.2493
15x15C02	HI-STAR	0.9297	0.9255	0.0007	0.2457
15x15C03	HI-STAR	0.9297	0.9255	0.0007	0.2440
15x15C04	HI-STAR	0.9311	0.9268	0.0008	0.2435
B15x15C01	HI-TRAC	0.9362	0.9319	0.0008	0.2374
B15x15C01	HI-STAR	0.9361	0.9316	0.0009	0.2385

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

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
15x15D01	HI-STAR	0.9341	0.9298	0.0008	0.2822
15x15D02	HI-STAR	0.9367	0.9324	0.0008	0.2802
15x15D03	HI-STAR	0.9354	0.9311	0.0008	0.2844
15x15D04	HI-TRAC	0.9354	0.9309	0.0009	0.2963
15x15D04	HI-STAR	0.9339	0.9292	0.0010	0.2958
15x15E01	HI-TRAC	0.9392	0.9349	0.0008	0.2827
15x15E01	HI-STAR	0.9368	0.9325	0.0008	0.2826
15x15F01	HI-STORM	0.3648	0.3614	0.0003	3.03E+04
15x15F01	HI-TRAC	0.9393	0.9347	0.0009	0.2925
15x15F01	HI-STAR	0.9395	0.9350	0.0009	0.2903
15x15G01	HI-TRAC	0.8878	0.8836	0.0007	0.3347
15x15G01	HI-STAR	0.8876	0.8833	0.0008	0.3357
15x15H01	HI-TRAC	0.9333	0.9288	0.0009	0.2353
15x15H01	HI-STAR	0.9337	0.9292	0.0009	0.2349
~~16x16A01~~	~~HI-STORM~~	~~0.3447~~	~~0.3412~~	~~0.0004~~	~~3.15E+04~~
~~16x16A01~~	~~HI-TRAC~~	~~0.9273~~	~~0.9228~~	~~0.0009~~	~~0.2710~~
16x16A01	HI-STAR	0.9287	0.9244	0.0008	0.2704
16x16A02	HI-STAR	0.9263	0.9221	0.0007	0.2702
16x16A03	*HI-STORM*	*0.3588*	*0.3555*	*0.0003*	*2.11E+04*
16x16A03	*HI-TRAC*	*0.9322*	*0.9278*	*0.0008*	*0.2673*
16x16A03	*HI-STAR*	*0.9327*	*0.9282*	*0.0009*	*0.2661*
17x17A01	HI-STORM	0.3243	0.3210	0.0003	3.23E+04

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

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
17x17A01	HI-TRAC	0.9378	0.9335	0.0008	0.2133
17x17A01	HI-STAR	0.9368	0.9325	0.0008	0.2131
17x17A02	HI-STAR	0.9329	0.9286	0.0008	0.2018
17x17B01	HI-STAR	0.9288	0.9243	0.0009	0.2607
17x17B02	HI-STAR	0.9290	0.9247	0.0008	0.2596
17x17B03	HI-STAR	0.9243	0.9199	0.0008	0.2625
17x17B04	HI-STAR	0.9324	0.9279	0.0009	0.2576
17x17B05	HI-STAR	0.9266	0.9222	0.0008	0.2539
17x17B06	HI-TRAC	0.9318	0.9275	0.0008	0.2570
17x17B06	HI-STAR	0.9311	0.9268	0.0008	0.2593
17x17C01	HI-STAR	0.9293	0.9250	0.0008	0.2595
17x17C02	HI-TRAC	0.9319	0.9274	0.0009	0.2610
17x17C02	HI-STAR	0.9336	0.9293	0.0008	0.2624

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6A01	HI-STAR	0.7539	0.7498	0.0007	0.2754
6x6A02	HI-STAR	0.7517	0.7476	0.0007	0.2510
6x6A03	HI-STAR	0.7545	0.7501	0.0008	0.2494
6x6A04	HI-STAR	0.7537	0.7494	0.0008	0.2494
6x6A05	HI-STAR	0.7555	0.7512	0.0008	0.2470
6x6A06	HI-STAR	0.7618	0.7576	0.0008	0.2298
6x6A07	HI-STAR	0.7588	0.7550	0.0005	0.2360
6x6A08	HI-STAR	0.7808	0.7766	0.0007	0.2527
B6x6A01	HI-TRAC	0.7732	0.7691	0.0007	0.2458
B6x6A01	HI-STAR	0.7727	0.7685	0.0007	0.2460
B6x6A02	HI-TRAC	0.7785	0.7741	0.0008	0.2411
B6x6A02	HI-STAR	0.7782	0.7738	0.0008	0.2408
B6x6A03	HI-TRAC	0.7886	0.7846	0.0007	0.2311
B6x6A03	HI-STAR	0.7888	0.7846	0.0007	0.2310
6x6B01	HI-STAR	0.7604	0.7563	0.0007	0.2461
6x6B02	HI-STAR	0.7618	0.7577	0.0007	0.2450
6x6B03	HI-STAR	0.7619	0.7578	0.0007	0.2439
6x6B04	HI-STAR	0.7686	0.7644	0.0008	0.2286
6x6B05	HI-STAR	0.7824	0.7785	0.0006	0.2184
B6x6B01	HI-TRAC	0.7833	0.7794	0.0006	0.2181
B6x6B01	HI-STAR	0.7822	0.7783	0.0006	0.2190

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6C01	HI-STORM	0.2759	0.2726	0.0003	1.59E+04
6x6C01	HI-TRAC	0.8024	0.7982	0.0008	0.2135
6x6C01	HI-STAR	0.8021	0.7980	0.0007	0.2139
7x7A01	HI-TRAC	0.7963	0.7922	0.0007	0.2016
7x7A01	HI-STAR	0.7974	0.7932	0.0008	0.2015
7x7B01	HI-STAR	0.9372	0.9330	0.0007	0.3658
7x7B02	HI-STAR	0.9301	0.9260	0.0007	0.3524
7x7B03	HI-STAR	0.9313	0.9271	0.0008	0.3438
7x7B04	HI-STAR	0.9311	0.9270	0.0007	0.3816
7x7B05	HI-STAR	0.9350	0.9306	0.0008	0.3382
7x7B06	HI-STAR	0.9298	0.9260	0.0006	0.3957
B7x7B01	HI-TRAC	0.9367	0.9324	0.0008	0.3899
B7x7B01	HI-STAR	0.9375	0.9332	0.0008	0.3887
B7x7B02	HI-STORM	0.4061	0.4027	0.0003	2.069E+04
B7x7B02	HI-TRAC	0.9385	0.9342	0.0008	0.3952
B7x7B02	HI-STAR	0.9386	0.9344	0.0007	0.3983
8x8A01	HI-TRAC	0.7662	0.7620	0.0008	0.2250
8x8A01	HI-STAR	0.7685	0.7644	0.0007	0.2227
8x8A02	HI-TRAC	0.7690	0.7650	0.0007	0.2163
8x8A02	HI-STAR	0.7697	0.7656	0.0007	0.2158
8x8B01	HI-STAR	0.9310	0.9265	0.0009	0.2935
8x8B02	HI-STAR	0.9227	0.9185	0.0007	0.2993

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
8x8B03	HI-STAR	0.9299	0.9257	0.0008	0.3319
8x8B04	HI-STAR	0.9236	0.9194	0.0008	0.3700
B8x8B01	HI-TRAC	0.9352	0.9310	0.0008	0.3393
B8x8B01	HI-STAR	0.9346	0.9301	0.0009	0.3389
B8x8B02	HI-TRAC	0.9401	0.9359	0.0007	0.3331
B8x8B02	HI-STAR	0.9385	0.9343	0.0008	0.3329
B8x8B03	HI-STORM	0.3934	0.3900	0.0004	1.815E+04
B8x8B03	HI-TRAC	0.9427	0.9385	0.0008	0.3278
B8x8B03	HI-STAR	0.9416	0.9375	0.0007	0.3293
8x8C01	HI-STAR	0.9315	0.9273	0.0007	0.2822
8x8C02	HI-STAR	0.9313	0.9268	0.0009	0.2716
8x8C03	HI-STAR	0.9329	0.9286	0.0008	0.2877
8x8C04	HI-STAR	0.9348	0.9307	0.0007	0.2915
8x8C05	HI-STAR	0.9353	0.9312	0.0007	0.2971
8x8C06	HI-STAR	0.9353	0.9312	0.0007	0.2944
8x8C07	HI-STAR	0.9314	0.9273	0.0007	0.2972
8x8C08	HI-STAR	0.9339	0.9298	0.0007	0.2915
8x8C09	HI-STAR	0.9301	0.9260	0.0007	0.3183
8x8C10	HI-STAR	0.9317	0.9275	0.0008	0.3018
8x8C11	HI-STAR	0.9328	0.9287	0.0007	0.3001
8x8C12	HI-STAR	0.9285	0.9242	0.0008	0.3062
B8x8C01	HI-TRAC	0.9348	0.9305	0.0008	0.3114

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
B8x8C01	HI-STAR	0.9357	0.9313	0.0009	0.3141
B8x8C02	HI-STORM	0.3714	0.3679	0.0004	2.30E+04
B8x8C02	HI-TRAC	0.9402	0.9360	0.0008	0.3072
B8x8C02	HI-STAR	0.9425	0.9384	0.0007	0.3081
B8x8C03	HI-TRAC	0.9429	0.9386	0.0008	0.3045
B8x8C03	HI-STAR	0.9418	0.9375	0.0008	0.3056
8x8D01	HI-STAR	0.9342	0.9302	0.0006	0.2733
8x8D02	HI-STAR	0.9325	0.9284	0.0007	0.2750
8x8D03	HI-STAR	0.9351	0.9309	0.0008	0.2731
8x8D04	HI-STAR	0.9338	0.9296	0.0007	0.2727
8x8D05	HI-STAR	0.9339	0.9294	0.0009	0.2700
8x8D06	HI-STAR	0.9365	0.9324	0.0007	0.2777
8x8D07	HI-STAR	0.9341	0.9297	0.0009	0.2694
8x8D08	HI-STAR	0.9376	0.9332	0.0009	0.2841
B8x8D01	HI-TRAC	0.9408	0.9368	0.0006	0.2773
B8x8D01	HI-STAR	0.9403	0.9363	0.0007	0.2778
8x8E01	HI-TRAC	0.9309	0.9266	0.0008	0.2834
8x8E01	HI-STAR	0.9312	0.9270	0.0008	0.2831
8x8F01	HI-TRAC	0.9396	0.9356	0.0006	0.2255
8x8F01	HI-STAR	0.9411	0.9366	0.0009	0.2264
9x9A01	HI-STAR	0.9353	0.9310	0.0008	0.2875
9x9A02	HI-STAR	0.9388	0.9345	0.0008	0.2228

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
9x9A03	HI-STAR	0.9351	0.9310	0.0007	0.2837
9x9A04	HI-STAR	0.9396	0.9355	0.0007	0.2262
B9x9A01	HI-STORM	0.3365	0.3331	0.0003	1.78E+04
B9x9A01	HI-TRAC	0.9434	0.9392	0.0007	0.2232
B9x9A01	HI-STAR	0.9417	0.9374	0.0008	0.2236
9x9B01	HI-STAR	0.9380	0.9336	0.0008	0.2576
9x9B02	HI-STAR	0.9373	0.9329	0.0009	0.2578
9x9B03	HI-STAR	0.9417	0.9374	0.0008	0.2545
B9x9B01	HI-TRAC	0.9417	0.9376	0.0007	0.2504
B9x9B01	HI-STAR	0.9436	0.9394	0.0008	0.2506
9x9C01	HI-TRAC	0.9377	0.9335	0.0008	0.2697
9x9C01	HI-STAR	0.9395	0.9352	0.0008	0.2698
9x9D01	HI-TRAC	0.9387	0.9343	0.0008	0.2635
9x9D01	HI-STAR	0.9394	0.9350	0.0009	0.2625
9x9E01	HI-STAR	0.9334	0.9293	0.0007	0.2227
9x9E02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9E02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9E02	HI-STAR	0.9401	0.9359	0.0008	0.2065
9x9F01	HI-STAR	0.9307	0.9265	0.0007	0.2899
9x9F02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9F02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9F02	HI-STAR	0.9401	0.9359	0.0008	0.2065

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

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
9x9G01	HI-TRAC	0.9307	0.9265	0.0007	0.2193
9x9G01	HI-STAR	0.9309	0.9265	0.0008	0.2191
10x10A01	HI-STAR	0.9377	0.9335	0.0008	0.3170
10x10A02	HI-STAR	0.9426	0.9386	0.0007	0.2159
10x10A03	HI-STAR	0.9396	0.9356	0.0007	0.3169
B10x10A01	HI-STORM	0.3379	0.3345	0.0003	1.74E+04
B10x10A01	HI-TRAC	0.9448	0.9405	0.0008	0.2214
B10x10A01	HI-STAR	0.9457	0.9414	0.0008	0.2212
10x10B01	HI-STAR	0.9384	0.9341	0.0008	0.2881
10x10B02	HI-STAR	0.9416	0.9373	0.0008	0.2333
10x10B03	HI-STAR	0.9375	0.9334	0.0007	0.2856
B10x10B01	HI-TRAC	0.9443	0.9401	0.0007	0.2380
B10x10B01	HI-STAR	0.9436	0.9395	0.0007	0.2366
10x10C01	HI-TRAC	0.9430	0.9387	0.0008	0.2424
10x10C01	HI-STAR	0.9433	0.9392	0.0007	0.2416
10x10D01	HI-TRAC	0.9383	0.9343	0.0007	0.3359
10x10D01	HI-STAR	0.9376	0.9333	0.0008	0.3355
10x10E01	HI-TRAC	0.9157	0.9116	0.0007	0.3301
10x10E01	HI-STAR	0.9185	0.9144	0.0007	0.2936

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

MPC-24 400PPM SOLUBLE BORON					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.8884	0.8841	0.0008	0.2501
B14x14B01	HI-STAR	0.8900	0.8855	0.0009	0.3173
14x14C03	HI-STAR	0.8950	0.8907	0.0008	0.3410
14x14D01	HI-STAR	0.8518	0.8475	0.0008	0.4395
14x14E02	HI-STAR	0.7132	0.7090	0.0007	0.4377
15x15A01	HI-STAR	0.9119	0.9076	0.0008	0.3363
B15x15B01	HI-STAR	0.9284	0.9241	0.0008	0.3398
B15x15C01	HI-STAR	0.9236	0.9193	0.0008	0.3074
15x15D04	HI-STAR	0.9261	0.9218	0.0008	0.3841
15x15E01	HI-STAR	0.9265	0.9221	0.0008	0.3656
15x15F01	HI-STORM (DRY)	0.4013	0.3978	0.0004	28685
15x15F01	HI-TRAC	0.9301	0.9256	0.0009	0.3790
15x15F01	HI-STAR	0.9314	0.9271	0.0008	0.3791
15x15G01	HI-STAR	0.8939	0.8897	0.0007	0.4392
15x15H01	HI-TRAC	0.9345	0.9301	0.0008	0.3183
15x15H01	HI-STAR	0.9366	0.9320	0.0009	0.3175
16x16A03~~1~~	HI-STAR	0.899~~355~~	0.894~~812~~	0.0009~~8~~	0.3164~~227~~
17x17A01	HI-STAR	0.9264	0.9221	0.0008	0.2801
17x17B06	HI-STAR	0.9284	0.9241	0.0008	0.3383
17x17C02	HI-TRAC	0.9296	0.9250	0.0009	0.3447
17x17C02	HI-STAR	0.9294	0.9249	0.0009	0.3433

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

MPC-24E/MPC-24EF, UNBORATED WATER					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.9380	0.9337	0.0008	0.2277
B14x14B01	HI-STAR	0.9312	0.9269	0.0008	0.2927
14x14C01	HI-STAR	0.9356	0.9311	0.0009	0.3161
14x14D01	HI-STAR	0.8875	0.8830	0.0009	0.4026
14x14E02	HI-STAR	0.7651	0.7610	0.0007	0.3645
15x15A01	HI-STAR	0.9336	0.9292	0.0008	0.2879
B15x15B01	HI-STAR	0.9465	0.9421	0.0008	0.2924
B15x15C01	HI-STAR	0.9462	0.9419	0.0008	0.2631
15x15D04	HI-STAR	0.9440	0.9395	0.0009	0.3316
15x15E01	HI-STAR	0.9455	0.9411	0.0009	0.3178
15x15F01	HI-STORM (DRY)	0.3699	0.3665	0.0004	3.280e+04
15x15F01	HI-TRAC	0.9465	0.9421	0.0009	0.3297
15x15F01	HI-STAR	0.9468	0.9424	0.0008	0.3270
15x15G01	HI-STAR	0.9054	0.9012	0.0007	0.3781
15x15H01	HI-STAR	0.9423	0.9381	0.0008	0.2628
16x16A03~~1~~	HI-STAR	0.93*94*~~41~~	0.9*351*~~297~~	0.000~~8~~*9*	~~0.3019~~*NC*
17x17A01	HI-TRAC	0.9467	0.9425	0.0008	0.2372
17x17A01	HI-STAR	0.9447	0.9406	0.0007	0.2374
17x17B06	HI-STAR	0.9421	0.9377	0.0008	0.2888
17x17C02	HI-STAR	0.9433	0.9390	0.0008	0.2932

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

MPC-24E/MPC-24EF, 300PPM BORATED WATER					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.8963	0.8921	0.0008	0.2231
B14x14B01	HI-STAR	0.8974	0.8931	0.0008	0.3214
14x14C01	HI-STAR	0.9031	0.8988	0.0008	0.3445
14x14D01	HI-STAR	0.8588	0.8546	0.0007	0.4407
14x14E02	HI-STAR	0.7249	0.7205	0.0008	0.4186
15x15A01	HI-STAR	0.9161	0.9118	0.0008	0.3408
B15x15B01	HI-STAR	0.9321	0.9278	0.0008	0.3447
B15x15C01	HI-STAR	0.9271	0.9227	0.0008	0.3121
15x15D04	HI-STAR	0.9290	0.9246	0.0009	0.3950
15x15E01	HI-STAR	0.9309	0.9265	0.0009	0.3754
15x15F01	HI-STORM (DRY)	0.3897	0.3863	0.0003	3.192E+04
15x15F01	HI-TRAC	0.9333	0.9290	0.0008	0.3900
15x15F01	HI-STAR	0.9332	0.9289	0.0008	0.3861
15x15G01	HI-STAR	0.8972	0.8930	0.0007	0.4473
15x15H01	HI-TRAC	0.9399	0.9356	0.0008	0.3235
15x15H01	HI-STAR	0.9399	0.9357	0.0008	0.3248
16x16A03~~1~~	HI-STAR	0.*9060*~~21~~	0.*9017*~~8977~~	0.000*8*~~9~~	~~0.3274~~*NC*
17x17A01	HI-STAR	0.9332	0.9287	0.0009	0.2821
17x17B06	HI-STAR	0.9316	0.9273	0.0008	0.3455
17x17C02	HI-TRAC	0.9320	0.9277	0.0008	0.2819
17x17C02	HI-STAR	0.9312	0.9270	0.0007	0.3530

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

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
16x16A0~~1~~	HI-STAR	0.94~~68~~	0.94~~27~~	0.0007	0.4488~~3925~~
17x17A01	HI-STAR	0.9111	0.9072	0.0006	0.4055
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365
17x17C02	HI-TRAC	0.9365	0.9327	0.0006	0.4468
17x17C02	HI-STAR	0.9355	0.9317	0.0006	0.4469

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

MPC-32, 5.0% Enrichment, Bounding Cases					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.9000	0.8959	0.0007	0.4651
B14x14B01	HI-STAR	0.9214	0.9175	0.0006	0.6009
14x14C01	HI-STAR	0.9480	0.9440	0.0006	0.6431
14x14D01	HI-STAR	0.9050	0.9009	0.0007	0.7276
14x14E02	HI-STAR	0.7415	0.7375	0.0006	0.9226
15x15A01	HI-STAR	0.9230	0.9189	0.0007	0.7143
B15x15B01	HI-STAR	0.9429	0.9390	0.0006	0.7234
B15x15C01	HI-STAR	0.9307	0.9268	0.0006	0.6439
15x15D04	HI-STAR	0.9466	0.9425	0.0007	0.7525
15x15E01	HI-STAR	0.9434	0.9394	0.0007	0.7215
15x15F01	HI-STORM (DRY)	0.5142	0.5108	0.0004	1.228E+04
15x15F01	HI-TRAC	0.9470	0.9431	0.0006	0.7456
15x15F01	HI-STAR	0.9483	0.9443	0.0007	0.7426
15x15G01	HI-STAR	0.9251	0.9212	0.0006	0.9303
15x15H01	HI-STAR	0.9333	0.9292	0.0007	0.7015
16x16A03~~1~~	HI-STAR	0.94*29*~~74~~	0.9*388*~~434~~	0.000*7*~~6~~	0.59*20*~~36~~
17x17A01	HI-STAR	0.9161	0.9122	0.0006	0.6141
17x17B06	HI-STAR	0.9371	0.9331	0.0006	0.6705
17x17C02	HI-TRAC	0.9436	0.9396	0.0006	0.6773
17x17C02	HI-STAR	0.9437	0.9399	0.0006	0.6780

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

Note: Maximum k_{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.I

CRITICALITY EVALUATION OF THE HI-STORM 100U SYSTEM

6.I.0 INTRODUCTION

This supplement is solely focused on providing an evaluation of criticality safety of the HI-STORM 100U system. The evaluation presented herein supplements those evaluations of the HI-STORM 100 overpack contained in the main body of Chapter 6 of this FSAR.

6.I.1 ACCEPTANCE CRITERIA

The acceptance criteria for criticality evaluations for the HI-STORM 100U system are identical to the criteria for the HI-STORM 100 system analyzed in the main body of Chapter 6.

6.I.2 EVALUATION

The HI-STORM 100U system differs from the HI-STORM 100 system only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs and HI-TRAC transfer casks are identical between the two systems.

The limiting condition in the main body of Chapter 6 of this FSAR is the fully flooded MPC in a HI-TRAC. This limiting condition is unaffected by the different storage overpacks. All results and conclusions for the fully flooded condition from the main body of Chapter 6 are therefore directly applicable to the HI-STORM 100U system.

During storage conditions, the maximum k_{eff} is significantly below the limiting maximum k_{eff} since the MPC is internally dry (see Section 6.1). Under this condition, the configuration is very similar between the HI-STORM 100U VVM and the HI-STORM 100 overpack analyzed in the main body of Chapter 6, consisting of the internally dry MPC, the gap between the MPC and the overpack, a steel shell or shells and concrete (above-ground) or soil (below-ground).Results for the VVM would therefore be practically identical to the results listed for storage conditions in the main body of Chapter 6. Any small differences in results would not affect the principal conclusions, since the maximum k_{eff} under storage conditions is so low.

In summary, the limiting condition for the HI-STORM 100U is identical to the limiting condition for the HI-STORM 100, and other conditions are very similar between the two systems from a criticality perspective. All results and conclusions for the HI-STORM 100 system presented in the main body of Chapter 6 are therefore directly applicable to the HI-STORM 100U system, and no additional calculations to demonstrate criticality safety are required for the HI-STORM 100U system.

7.1 CONFINEMENT BOUNDARY

The primary confinement boundary against the release of radionuclides is the cladding of the individual fuel rods. The spent fuel rods are protected from degradation by maintaining an inert gas atmosphere (helium) inside the MPC and keeping the fuel cladding temperatures below the design basis values specified in Chapter 2.

The HI-STORM 100 confinement boundary consists of any one of the fully-welded MPC designs described in Chapter 1. Each MPC is identical from a confinement perspective so the following discussion applies to all MPCs. The confinement boundary of the MPC consists of:

- MPC shell
- bottom baseplate
- MPC lid (including the vent and drain port cover plates)
- MPC closure ring
- associated welds

The above items form a totally seal-welded vessel for the storage of design basis spent fuel assemblies.

The MPC requires no valves, gaskets or mechanical seals for confinement. Figure 7.1.1 shows an elevation cross-section of the MPC confinement boundary. All components of the confinement boundary are Important to Safety, Category A, as specified in Table 2.2.6. The MPC confinement boundary is designed and fabricated in accordance with the ASME Code, Section III, Subsection NB [7.1.1] to the maximum extent practicable. Chapter 2 provides design criteria for the confinement design. Section 2.2.4 provides applicable Code requirements. NRC-approved alternatives to specific Code requirements with complete justifications are presented in Table 2.2.15.

7.1.1 Confinement Vessel

The HI-STORM 100 System confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The MPC is designed, fabricated, inspected, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], including certain NRC-approved alternatives. The MPC shell and baseplate assembly and basket structure are delivered to the loading facility as one complete component. The MPC lid, vent and drain port cover plates, and closure ring are supplied separately and are installed following fuel loading. The MPC lid and closure ring are welded to the upper part of the MPC shell after fuel loading to provide redundant sealing of the confinement boundary. The vent and drain port cover plates are welded to the MPC lid after the lid is welded to the MPC. The welds forming the confinement boundary are described in detail in Section 7.1.3.

The MPC lid is made intentionally thick to minimize radiation exposure to workers during MPC closure operations, and is welded to the MPC shell. The vent and drain port cover plates are welded to the MPC lid following completion of MPC draining, moisture removal, and helium backfill activities to close the MPC vent and drain openings. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter. The combination of the welded MPC lid and closure ring form the redundant closure of the MPC.

Table 7.1.1 provides a summary of the design ratings for normal, off-normal and accident conditions for the MPC confinement vessel. Tables 1.2.2, 2.2.1, and 2.2.3 provide additional design basis information.

The MPC shell and baseplate are helium leakage tested during fabrication in accordance with the requirements defined in Chapter 9. Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by liquid penetrant method, volumetrically examined (or, if volumetric examination is not performed, multi-layer liquid penetrant examination must be performed), and pressure tested. If the MPC lid weld is acceptable, the vent and drain port cover plates are welded in place and examined by the liquid penetrant method. Finally, the MPC closure ring is installed, welded and inspected by the liquid penetrant method. Chapters 8, 9, and 12 provide procedural guidance, acceptance criteria, and operating controls, respectively, for performance and acceptance of liquid penetrant examinations, volumetric examination, and pressure testing of the field welds on the MPC.

After moisture removal, the MPC cavity is backfilled with helium. The helium backfill provides an inert atmosphere within the MPC cavity that precludes oxidation and hydride attack of the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and reduces the maximum fuel cladding temperatures. MPC inerting, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

7.1.2 Confinement Penetrations

The MPC penetrations are designed to prevent the release of radionuclides under all normal, off-normal and accident conditions of storage. Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and for fuel cool-down and MPC flooding during unloading operations. No other confinement penetrations exist in the MPC. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain connectors allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The MPC vent and drain ports are sealed by cover plates that are seal welded to the MPC lid. No credit is taken for the seal provided by the vent and drain port caps. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, providing the redundant closure of the MPC vessel. The redundant closures of the MPC satisfy the requirements of 10CFR72.236(e) [7.0.1].

The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection.

7.1.3 Seals and Welds

The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], with certain NRC-approved alternatives. The MPC has no bolted closures or mechanical seals. Section 7.1.1 describes the design of the confinement vessel welds. The welds forming the confinement boundary are summarized in Table 7.1.2.

Confinement boundary welds are performed, inspected, and tested in accordance with the applicable requirements of ASME Section III, Subsection NB [7.1.1] with certain NRC-approved alternatives. The use of multi-pass welds, root pass, for multiple pass welds, and final surface liquid penetrant inspection, and volumetric examination essentially eliminates the chance of a pinhole leak through the weld. If volumetric examination is not performed, multi-layer liquid penetrant examination must be performed. Welds other than the field closure welds are also helium leak tested in the fabrication shop, providing added assurance of weld integrity. Additionally, a Code pressure test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity after fuel loading. The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld does not result in release of radioactive material to the environment. Table 9.1.4 provides a summary of the closure weld examinations and tests.

7.1.4 Closure

The MPC is a totally seal-welded pressure vessel. The MPC has no bolted closure or mechanical seals. The MPC's redundant closures are designed to maintain confinement integrity during normal conditions of storage, and off-normal and postulated accident conditions. There are no unique or special closure devices. Primary closure welds (lid-to-shell and vent/drain port cover plate-to-lid) are examined using the liquid penetrant technique to ensure their integrity. A description of the MPC weld examinations is provided in Chapter 9.

Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required. Chapter 11 describes requirements for verifying the continued confinement capabilities of the MPC in the event of off-normal or accident conditions. As discussed in Section 2.3.3.2, no instrumentation is required or provided for HI-STORM 100 System storage operations, other than normal security service instruments and TLDs.

7.1.5 Damaged Fuel Container

The MPC is designed to allow for the storage of specified damaged fuel assemblies and fuel debris in a specially designed damaged fuel container (DFC). Fuel assemblies classified as damaged fuel or fuel debris as specified in Section 2.1.9 have been evaluated.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel DFCs for storage in the HI-STORM 100 System. The DFCs that may be loaded into the MPCs are discussed in Section 2.1.3. The DFC is designed to provide SNF loose component retention and handling capabilities. The DFC consists of a smooth-walled, welded stainless steel square container with a removable lid. The container lid provides the means of DFC closure and handling. The DFC is provided with stainless steel wire mesh screens in the top and bottom for draining, moisture removal and helium backfill operations. The screens are specified as a 250-by-250-mesh with an effective opening of 0.0024 inches. There are no other openings in the DFC. Section 2.1.9 specifies the fuel assembly characteristics for damaged fuel *and fuel debris* acceptable for loading in the MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F or MPC-68FF ~~and for fuel debris acceptable for loading in the MPC-24EF, MPC-32F, MPC-68F or MPC-68FF~~.

Since the DFC has screens on the top and bottom, the DFC provides no pressure retention function. The confinement function of the DFC is limited to minimizing the release of loose particulates within the sealed MPC. The confinement function of the MPC is not altered by the presence of the DFCs. The radioactive material available for release from the specified fuel assemblies are bounded by the design basis fuel assemblies analyzed herein.

7.1.6 Design and Qualification of Final MPC Closure Welds

The Holtec MPC final closure welds meet the criteria of NRC Interim Staff Guidance (ISG)18 [7.1.2] such that leakage from the MPC confinement boundary is not considered credible. Table 7.1.4 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one. In addition, because proper execution of the MPC lid-to-shell weld is vital to ensuring no credible leakage from the field-welded MPC, Holtec shall review the closure welding procedures for conformance to Code and FSAR requirements.

Table 7.1.1

SUMMARY OF CONFINEMENT BOUNDARY DESIGN SPECIFICATIONS

Design Condition	Design Pressure (psig)	Design Temperature (°F)
Normal	100	MPC Lid: 550
		MPC Shell: 500
		MPC Baseplate: 400
Off-Normal	110	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775
Accident	200	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775

Table 7.1.2

MPC CONFINEMENT BOUNDARY WELDS

Confinement Boundary Welds		
MPC Weld Location	**Weld Type†**	**ASME Code Category (Section III, Subsection NB)**
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port cover plates to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to closure ring radial	Partial Penetration Groove (field weld)	††
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C

† The tests and inspections for the confinement boundary welds are listed in Section 9.1.1.

†† This joint is governed by NB-5271 (liquid penetrant examination).

Table 7.1.3

TABLE DELETED

Table 7.1.4

COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE FOR STORAGE

DESIGN/QUALIFICATION GUIDANCE	HOLTEC MPC DESIGN	FSAR REFERENCE
The canister is constructed from austenitic stainless steel	The MPC enclosure vessel is constructed entirely from austenitic stainless steel (Alloy X). Alloy X is defined as Type 304, 304LN, 316, or 316LN material	Section 1.2.1.1 and Appendix 1.A
The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3	The MPC lid-to-shell (LTS) closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8" of weld depth, which corresponds to the critical flaw size. A weld quality factor of 0.45 (45% of actual weld capacity) has been used in the stress analysis.	Section 9.1.1.1 and Tables 2.2.15 and 9.1.4. HI-STAR FSAR Section 3.4.4.3.1.5 and Appendix 3.E (Docket 72-1008)
The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents, and natural phenomena	The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is design to withstand 45 g deceleration loadings and the cask system is analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g's.	Section 3.4.4.3 and Appendix 3.A. HI-STAR FSAR Section 3.4.4.3
Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10 CFR 72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.	Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STORM CoC. Compliance with 10 CFR 72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.	Section 9.1.1.1 and Table 2.2.15 Section 13.0
Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.	The NRC has approved the Holtec quality assurance program under 10 CFR 71. That QA program approval has been adopted for activities governed by 10 CFR 72 as permitted by 10 CFR 72.140(d)	Section 13.0

SUPPLEMENT 7.I

CONFINEMENT EVALUATION OF THE HI-STORM 100U SYSTEM

7.I.1 INTRODUCTION

This supplement is solely focused on providing an evaluation of confinement safety of the HI-STORM 100U System. The evaluation presented herein supplements those evaluations of the HI-STORM 100 and HI-STORM 100S overpacks contained in the main body of Chapter 7 of this FSAR.

7.I.2 ACCEPTANCE CRITERIA

The acceptance criteria for confinement evaluations for the HI-STORM 100U system are identical to the criteria for the HI-STORM 100 System analyzed in the main body of Chapter 7.

7.I.3 EVALUATION

The HI-STORM 100U System differs from the HI-STORM 100 System only in the use of a different storage overpack, the HI-STORM 100U vertical ventilated module (VVM). All MPCs are identical between the two systems.

In summary, as the MPC design is unchanged by the addition of the HI-STORM 100U VVM, the material presented in Section 7.1.6 certifying that the MPC meets the criteria of ISG-18, is valid for the HI-STORM 100U System. Therefore, leakage from the MPC confinement boundary is not considered credible and no confinement analysis is required.

CHAPTER 8: OPERATING PROCEDURES†

8.0 INTRODUCTION:

This chapter outlines the loading, unloading, and recovery procedures for the HI-STORM 100 System for storage operations. The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance is met and the requirements of the CoC are met. The information provided in this chapter meets all requirements of NUREG-1536 [8.0.1].

Section 8.1 provides the guidance for loading the HI-STORM 100 System in the spent fuel pool. Section 8.2 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 8.3 provides the procedure for unloading the HI-STORM 100 System in the spent fuel pool. Section 8.4 provides the guidance for MPC transfer to the HI-STAR 100 Overpack for transport or storage. Section 8.4 can also be used for recovery of a breached MPC for transport or storage. Section 8.5 provides the guidance for transfer of the MPC into HI-STORM from the HI-STAR 100 transport overpack. *Supplementary guidance for each of the aforementioned sections that are specific to HI-STORM 100U operations are found in Supplement 8.I.* Equipment specific operating details such as Vacuum Drying System, valve manipulation and Transporter operation are not within the scope of this FSAR and will be provided to users based on the specific equipment selected by the users and the configuration of the site.

The procedures contained herein describe acceptable methods for performing HI-STORM 100 loading and unloading operations. Unless otherwise stated, references to the HI-STORM 100 apply equally to the HI-STORM 100, 100S ~~and~~ 100S Version B, *and 100U.* Users may alter these procedures to allow alternate methods and operations to be performed in parallel or out of sequence as long as the general intent of the procedure is met. In the figures following each section, acceptable configurations of rigging, piping, and instrumentation are shown. In some cases, the figures are artists renditions. Users may select alternate configurations, equipment and methodology to accommodate their specific needs provided that the intent of this guidance is met and the requirements of the CoC are met. All rigging should be approved by the user's load handling authority prior to use. User-developed procedures and the design and operation of any alternate equipment must be reviewed by the Certificate holder prior to implementation.

Licensees (Users) will utilize the procedures provided in this chapter, equipment-specific operating instructions, and plant working procedures and apply them to develop the site specific written, loading and unloading procedures.

The loading and unloading procedures in Section 8.1 and 8.3 can also be appropriately revised

† 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).

into written site-specific procedures to allow dry loading and unloading of the system in a hot cell or other remote handling facility. The Dry Transfer Facility (DTF) loading and unloading procedures are essentially the same with respect to loading removing moisture, and inerting, of the MPC. The dry transfer facility shall develop the appropriate site-specific procedures as part of the DTF facility license.

Tables 8.1.1 through 8.1.4 provide the handling weights for each of the HI-STORM 100 System major components and the loads to be lifted during various phases of the operation of the HI-STORM 100 System. Users shall take appropriate actions to ensure that the lift weights do not exceed user-supplied lifting equipment rated loads. Table 8.1.5 provides the HI-STORM 100 System bolt torque and sequencing requirements. Table 8.1.6 provides an operational description of the HI-STORM 100 System ancillary equipment along with its safety designation, where applicable. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in the Certificate of Compliance and as defined in Section 2.1.9 are loaded into the HI-STORM 100 System.

In addition to the requirements set forth in the CoC, users will be required to develop or modify existing programs and procedures to account for the operation of an ISFSI. Written procedures will be required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of items and components identified as Important to Safety, 10CFR72.48 [8.1.1] programs, specialized instrument calibration, special nuclear material accountability at the ISFSI, security modifications, fuel handling procedures, training and emergency response, equipment and process qualifications. Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. Users are also required to take necessary actions to prevent the fuel cladding from exceeding temperature limits during drying operations and during handling of the MPC in the HI-TRAC transfer cask. Section 4.5 of the FSAR provides requirements on the necessary actions, if any, based on the heat load of the MPC.

Table 8.1.7 summarizes some of the instrumentation used to load and unload the HI-STORM 100 System. Tables 8.1.8, 8.1.9, and 8.1.10 provide sample receipt inspection checklists for the HI-STORM 100 overpack, the MPC, and the HI-TRAC Transfer Cask, respectively. Users

Technical and Safety Basis for Loading and Unloading Procedures

The procedures herein (Sections 8.1.2 through 8.1.5) are developed for the loading, storage, unloading, and recovery of spent fuel in the HI-STORM 100 System. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present risks. The design of the HI-STORM 100 System, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize risks and mitigate consequences of potential events. To summarize, consideration is given in the loading and unloading systems and procedures to the potential events listed in Table 8.0.1.

The primary objective is to reduce the risk of occurrence and/or to mitigate the consequences of the event. The procedures contain Notes, Warnings, and Cautions to notify the operators to upcoming situations and provide additional information as needed. The Notes, Warnings and Cautions are purposely bolded and boxed and immediately precede the applicable steps.

In the event of an extreme abnormal condition (e.g., cask drop or tip-over event) the user shall have appropriate procedural guidance to respond to the situation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution and reporting to the appropriate regulatory agencies, as required.

Table 8.0.1
OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6. Procedural guidance is given for cask handling, inspection of lifting equipment, and proper engagement to the trunnions.	See Section 8.1.2.
Cask Tip-Over Prior to welding of the MPC lid	The Lid Retention System is available to secure the MPC lid during movement between the spent fuel pool and the cask preparation area.	See Section 8.1.5. See Figure 8.1.15.
Contamination of the MPC external shell	The annulus seal, pool lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water.	See Figures 8.1.13 and 8.1.14.
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.	See Figures 8.1.19-8.1.22.
Damage to fuel assembly cladding from oxidation/thermal shock	Fuel assemblies are never subjected to air or oxygen during loading and unloading operations. Cool-Down System brings fuel assembly bulk temperatures to below water boiling temperature prior to flooding.	See Section 8.1.5, and Section 8.3.3
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum Drying System is separate from pressurized gas and water systems.	See Figure 8.1.22 and 8.1.23.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to, and during welding or cutting activities. The space below the MPC lid shall be evacuated or purged prior to, and during these activities.	See Section 8.1.5 and Section 8.3.3.

Table 8.0.1
OPERATIONAL CONSIDERATIONS (CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Excess dose from failed fuel assemblies	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.	See Figure 8.1.16 and Section 8.3.3.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.	See ALARA Notes and Warnings throughout the procedures.
Excess generation of radioactive waste	The HI-STORM system uses process systems that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Where possible, items are installed by hand and require no tools.	Examples: HI-TRAC bottom protective cover, bolt plugs in empty holes, pre-wetting of components.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.	See Section 8.1.4.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in stages to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. If the forced helium dehydration process used, the temperature of the gas exiting the demoisturizer is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations	See Section 8.1.5

Table 8.0.1
OPERATIONAL CONSIDERATIONS (CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.	See Section 8.1.5.
Load Drop	Rigging diagrams and procedural guidance are provided for all lifts. Component weights are provided in Tables 8.1.1 through 8.1.4.	See Figures 8.1.6, 8.1.7, 8.1.9, 8.1.25 and 8.1.27. See Tables 8.1.1 through 8.1.4.
Over-pressurization of MPC during loading and unloading	Pressure relief valves in the water and gas processing systems limit the MPC pressure to acceptable levels.	See Figures 8.1.20, 8.1.21, 8.1.23 and 8.3.4.
Overstressing MPC lift lugs from side loading	The MPC is upended using the upending frame.	See Figure 8.1.6 and Section 8.1.2.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts.	See Section 8.1.7 for example. See Tables 8.1.1 through 8.1.4.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.	See Section 8.1.5 and Section 8.3.3.
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.	See Section 8.1.3 and Section 8.1.5.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during moisture removal and helium backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.	See Figure 8.1.11 and 8.1.16. See Section 8.3.3.

8.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

8.1.1 Overview of Loading Operations:

The HI-STORM 100 System is used to load, transfer and store spent fuel. Specific steps are performed to prepare the HI-STORM 100 System for fuel loading, to load the fuel, to prepare the system for storage and to place it in storage at an ISFSI. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC and/or HI-STORM may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications as long as the lift height restrictions are met (lift height restrictions apply only to suspended forms of transport). Users shall develop detailed written procedures to control on-site transport operations. Section 8.1.2 provides the general procedures for rigging and handling of the HI-STORM overpack and HI-TRAC transfer cask. Figure 8.1.1 shows a general flow diagram of the HI-STORM loading operations.

Refer to the boxes of Figure 8.1.2 for the following description. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into HI-TRAC (Box 2). The annulus is filled with plant demineralized water† and the MPC is filled with either spent fuel pool water or plant demineralized water (borated as required) (Box 3). An inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC to prevent spent fuel pool water from contaminating the exterior surface of the MPC. HI-TRAC and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the optional Lid Retention System (Box 6). The lift yoke remotely engages to the HI-TRAC lifting trunnions to lift the HI-TRAC and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC from the spent fuel pool, the cask is removed from the spent fuel pool. If the Lid Retention System is being used, the HI-TRAC top lid bolts are installed to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

HI-TRAC is placed in the designated preparation area and the Lift Yoke and Lid Retention System (if utilized) are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The Temporary Shield Ring (if utilized) is installed and filled with water and the neutron shield jacket is filled with water (if drained). The inflatable annulus seal is removed, and the annulus shield (if utilized) is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-TRAC during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

† Users may substitute domestic water in each step where demineralized water is specified.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the automated welding system (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. An ultrasonic or multi-layer PT examination is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The water level is raised to the top of the MPC and a hydrostatic test followed by an additional liquid penetrant examination is performed on the MPC Lid-to-Shell weld to verify structural integrity. To calculate the helium backfill requirements for the MPC (if the backfill is based upon helium mass or volume measurements), the free volume inside the MPC must first be determined. This free volume may be determined by measuring the volume of water displaced or any other suitable means.

Depending upon the burn-up of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system or forced helium dehydration system. For MPCs without high burn-up fuel, the vacuum drying system may be connected to the MPC and used to remove all liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and vacuum drying system lines. The internal pressure is reduced to below 3 torr and held for 30 minutes to ensure that all liquid water is removed.

For high-burn-up fuel, or as an alternative for MPCs without high burn-up fuel, a forced helium dehydration system is utilized to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. The temperature of the gas exiting the system demoisturizer is maintained below 21 °F for a minimum of 30 minutes to ensure that all liquid water is removed

Following MPC moisture removal, the MPC is backfilled with a predetermined amount of helium gas. . If the MPC contains high burn-up fuel, then a Supplemental Cooling System (SCS) (if required) is connected to the HI-TRAC annulus prior to helium backfill and is used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits (See Figure 2.C.1). The helium backfill ensures adequate heat transfer during storage, and provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes (for multi-pass welds) (Box 10).

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield (if utilized) is removed and the remaining water in the annulus is drained. The Temporary Shield Ring (if utilized) is drained and removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and HI-TRAC dose rates are measured. HI-TRAC top lid[3] is installed and the bolts are torqued (Box 11). The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point on the MPC. MPC slings are installed between the MPC lift cleats and the lift yoke (Box 12).

If the HI-TRAC 125 is being used, the transfer lid is attached to the HI-TRAC as follows. The HI-TRAC is positioned above the transfer slide to prepare for bottom lid replacement. The transfer slide consists of an adjustable-height rolling carriage and a pair of channel tracks. The transfer slide supports the transfer step which is used to position the two lids at the same elevation and creates a tight seam between the two lids to eliminate radiation streaming. The overhead crane is shut down to prevent inadvertent operation. The transfer slide carriage is raised to support the pool lid while the bottom lid bolts are removed. The transfer slide then lowers the pool lid and replaces the pool lid with the transfer lid. The carriage is raised and the bottom lid bolts are replaced. The MPC lift cleats and slings support the MPC during the transfer operations. Following the transfer, the MPC slings are disconnected and HI-TRAC is positioned for MPC transfer into HI-STORM.

MPC transfer may be performed inside or outside the fuel building (Box 13). Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways (Box 14 and 15). *MPC transfer to the HI-STORM 100U is described in Supplement 8.I.* The empty HI-STORM overpack is inspected and positioned with the lid removed. Vent duct shield inserts[1] are installed in the HI-STORM exit vent ducts. The vent duct shield inserts prevent radiation streaming from the HI-STORM Overpack as the MPC is lowered past the exit vents. If the HI-TRAC 125D is used, the mating device is positioned on top of the HI-STORM. The HI-TRAC is placed on top of HI-STORM. An alignment device (or mating device in the case of HI-TRAC 125D) helps guide HI-TRAC during this operation[2]. The MPC may be lowered using the MPC downloader, the main crane hook or other similar devices. The MPC downloader (if used) may be attached to the HI-TRAC lid or mounted to the overhead lifting device. The MPC slings are attached to the MPC lift cleats.

If used, the SCS will be disconnected from the HI-TRAC and the HI-TRAC annulus drained, prior to transfer of the MPC from the HI-TRAC to the HI-STORM. If the transfer doors are used (i.e. not the HI-TRAC 125D), the MPC is raised slightly, the transfer lid door locking pins are removed and the doors are opened. If the HI-TRAC 125D is used, the pool lid is removed and the mating device drawer is opened. Optional trim plates may be installed on the top and bottom of both doors (or drawer for HI-TRAC 125D) and secured using hand clamps. The trim plates eliminate radiation streaming above and below the doors (drawer). The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. The trim plates are removed, the doors (or drawer) are closed. The empty HI-TRAC must be removed with the doors open when the HI-

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1 Vent duct shield inserts are only used on the HI-STORM 100.

2 The alignment guide may be configured in many different ways to accommodate the specific sites. See Table 8.1.6.

3 Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid

STORM 100S is used to prevent interference with the lift cleats and slings. HI-TRAC is removed from on top of HI-STORM. The MPC slings and MPC lift cleats are removed. Hole plugs are installed in the empty MPC lifting holes to fill the voids left by the lift cleat bolts. The alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (if used) are removed, and the HI-STORM lid is installed. The exit vent gamma shield cross plates, temperature elements (if used) and vent screens are installed.. The HI-STORM lid studs and nuts are installed. The HI-STORM is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM Overpack and HI-TRAC transfer cask may be moved using a number of methods as long as the lifting equipment requirements are met. For sites with high seismic conditions, the HI-STORM 100A is anchored to the ISFSI. Once located at the storage pad, the inlet vent gamma shield cross plates are installed and the shielding effectiveness test is performed. Finally, the temperature elements and their instrument connections are installed (if used), and the air temperature rise testing (if required) is performed to ensure that the system is functioning within its design parameters.

8.1.2 HI-TRAC and HI-STORM Receiving and Handling Operations

Note:
HI-TRAC may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for HI-TRAC and HI-STORM handling. Site-specific procedures shall specify the required operational sequences based on the handling configuration at the sites.

1. Vertical Handling of HI-TRAC:
   a. Verify that the lift yoke load test certifications are current.
   b. Visually inspect the lifting device (lift yoke or lift links) and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Replace or repair damaged components as necessary.
   c. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
   d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

Note:
Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements.
Warning:
When lifting the loaded HI-TRAC with only the pool lid, the HI-TRAC should be carried as low as practicable. This minimizes the dose rates due to radiation scattering from the floor. Personnel should remain clear of the area and the HI-TRAC should be placed in position as soon as practicable.

   e. Raise HI-TRAC and position it accordingly.
2. Upending of HI-TRAC in the Transfer Frame:
   a. Position HI-TRAC under the lifting device. Refer to Step 1, above.

b. If necessary, remove the missile shield from the HI-TRAC Transfer Frame. See Figure 8.1.4.

c. Verify that the lift yoke load test certifications are current.

d. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

e. Deleted.

f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.

g. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

h. Slowly rotate HI-TRAC to the vertical position keeping all rigging as close to vertical as practicable. See Figure 8.1.4.

i. If used, lift the pocket trunnions clear of the Transfer Frame rotation trunnions.

3. Downending of HI-TRAC in the Transfer Frame:

ALARA Warning:
A loaded HI-TRAC should only be downended with the transfer lid or other auxiliary shielding installed.

a. Position the Transfer Frame under the lifting device.

b. Verify that the lift yoke load test certifications are current.

c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

d. Deleted.

e. Deleted.

f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.

g. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.

h. Position the pocket trunnions to receive the Transfer Frame rotation trunnions. See Figure 8.1.4 (Not used for HI-TRAC 125D).

i. Slowly rotate HI-TRAC to the horizontal position keeping all rigging as close to vertical as practicable.

j. Disengage the lift yoke.

4. Horizontal Handling of HI-TRAC in the Transfer Frame:

a. Verify that the Transfer Frame is secured to the transport vehicle as necessary.

b. Downend HI-TRAC on the Transfer Frame per Step 3, if necessary.

c. If necessary, install the HI-TRAC missile Shield on the HI-STAR 100 Transfer Frame (See Figure 8.1.4).

5. Vertical Handling of HI-STORM *(Not applicable to HI-STORM 100U)*:

Note:
The HI-STORM 100 Overpack may be lifted with a special lifting device that engages the overpack anchor blocks with threaded studs and connects to a cask transporter, crane, or similar equipment. The device is designed in accordance with ANSI N14.6.

a. Visually inspect the HI-STORM lifting device for gouges, cracks, deformation or other indications of damage.

b. Visually inspect the transporter lifting attachments for gouges, cracks, deformation or other indications of damage..

c. If necessary, attach the transporter's lifting device to the transporter and HI-STORM..

d. Raise and position HI-STORM accordingly. See Figure 8.1.5.

6. Empty MPC Installation in HI-TRAC:

Note:
To avoid side loading the MPC lift lugs, the MPC must be upended in the MPC Upending Frame (or equivalent). See Figure 8.1.6.

a. If necessary, rinse off any road dirt with water. Remove any foreign objects from the MPC internals.

b. If necessary, upend the MPC as follows:

   1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

   2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 8.1.6.

   3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 8.1.6.

   4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device) (See Figure 8.1.6).

   5. Raise the MPC in the Upending Frame.

Warning:
The Upending Frame corner should be kept close to the ground during the upending process.

   6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.

7. When the MPC approaches the vertical orientation, tension on the lower slings may be released.

8. Place the MPC in a vertical orientation.

9. Disconnect the MPC straps and disconnect the rigging.

c. Install the MPC in HI-TRAC as follows:

1. Install the four point lift sling to the lift lugs inside the MPC. See Figure 8.1.7.

2. Raise and place the MPC inside HI-TRAC.

Note:
An alignment punch mark is provided on HI-TRAC and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring. See Figure 8.1.8.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside HI-TRAC. Disconnect the MPC rigging or the MPC lift rig.

8.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

Note:
Receipt inspection, installation of the empty MPC in the HI-TRAC, and lower fuel spacer installation may occur at any location or be performed at any time prior to complete submersion in the spent fuel pool as long as appropriate steps are taken to prevent contaminating the exterior of the MPC or interior of the HI-TRAC.

ALARA Note:
A bottom protective cover may be attached to HI-TRAC pool lid bottom. This will help prevent imbedding contaminated particles in HI-TRAC bottom surface and ease the decontamination effort.

1. Place HI-TRAC in the cask receiving area. Perform appropriate contamination and security surveillances, as required.

2. If necessary, remove HI-TRAC Top Lid by removing the top lid bolts and using the lift sling. See Figure 8.1.9 for rigging.

   a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.

   b. Perform a radiological survey of the inside of HI-TRAC to verify there is no residual contamination from previous uses of the cask.

3. Disconnect the rigging.

4. Store the Top Lid and bolts in a site-approved location.

5. If necessary, configure HI-TRAC with the pool lid as follows:

ALARA Warning:
The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

a. Inspect the seal on the pool lid for cuts, cracks, gaps and general condition. Replace the seal if necessary.

b. Remove the bottom lid bolts and store them temporarily.

c. Raise the empty HI-TRAC and position it on top of the pool lid.

d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.

e. Install the pool lid bolts. See Table 8.1.5 for torque requirements.

f. If necessary, thread the drain connector pipe to the pool lid.

g. Store the HI-TRAC Transfer Lid in a site-approved location.

6. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

7. Install the MPC inside HI-TRAC and place HI-TRAC in the designated preparation area. See Section 8.1.2.

Note:
Upper fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Upper fuel spacer installation may occur any time prior to MPC lid installation.

8. Install the upper fuel spacers in the MPC lid as follows:

Warning:
Never work under a suspended load.

a. Position the MPC lid on supports to allow access to the underside of the MPC lid.

b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Figure 8.1.10 and Table 8.1.5 for torque requirements.

c. Install threaded plugs in the MPC lid where and when spacers will not be installed, if necessary. See Table 8.1.5 for torque requirements.

9. At the user's discretion perform an MPC lid and closure ring fit test:

Note:
It may be necessary to perform the MPC installation and inspection in a location that has sufficient crane clearance to perform the operation.

a. Visually inspect the MPC lid rigging (See Figure 8.1.9).

b. At the user's discretion, raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid. See Figure 8.1.11.

Note:
The MPC Shell is relatively flexible compared to the MPC Lid and may create areas of local contact that impede Lid insertion in the Shell. Grinding of the MPC Lid below the minimum diameter on the drawing is permitted to alleviate interference with the MPC Shell in areas of localized contact. If the amount of material removed from the surface exceeds 1/8", the surface shall be examined by a liquid penetrant method (NB-2546). The weld prep for the Lid-to-Shell weld shall be maintained after grinding.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. See Figure 8.1.12. Install the MPC lid. Verify that the MPC lid fit and weld prep are in accordance with the design drawings.

ALARA Note:
The closure ring is installed by hand. Some grinding may be required on the closure ring to adjust the fit.

d. Install, align and fit-up the closure ring.

e. Verify that closure ring fit and weld prep are in accordance with the fabrication drawings or the approved design drawings.

f. Remove the closure ring, vent and drain port cover plates and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.

10. At the user's discretion, perform an MPC vent and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

Note:
Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used.

11. Install lower fuel spacers in the MPC (if necessary). See Figure 8.1.10.

12. Fill the MPC and annulus as follows:

a. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

Caution:
Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

b. Manually insert the inflatable annulus seal around the MPC. See Figure 8.1.13.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal.

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

ALARA Note:
Bolt plugs, placed in, or waterproof tape over empty bolt holes, reduce the time required for decontamination.

13. At the user's discretion, install HI-TRAC top lid bolt plugs and/or apply waterproof tape over any empty bolt holes.

ALARA Note:
Keeping the water level below the top of the MPC prevents splashing during handling.

14. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches below the top of the MPC shell. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements.

15. If necessary for plant crane capacity limitations, drain the water from the neutron shield jacket. See Tables 8.1.1 through 8.1.4 as applicable.

16. Place HI-TRAC in the spent fuel pool as follows:

ALARA Note:
The term "Spent Fuel Pool" is used generically to refer to the users designated cask loading location. The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

   a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-TRAC. See Figure 8.1.14.

   b. Verify spent fuel pool for boron concentration requirements in accordance with 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.
2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.
3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

8.1.5 MPC Closure

Note:
The user may elect to use the Lid Retention System (See Figure 8.1.15) to assist in the installation of the MPC lid and lift yoke, and to provide the means to secure the MPC lid in the event of a drop accident during loaded cask handling operations outside of the spent fuel pool. The user is responsible for evaluating the additional weight imposed on the cask, lift yoke, crane and floor prior to use. See Tables 8.1.1 through 8.1.4 as applicable. The following guidance describes installation of the MPC lid using the lift yoke. The MPC lid may also be installed separately. Depending on facility configuration, users may elect to perform MPC closure operations with the HI-TRAC partially submerged in the spent fuel pool. If opted, operations involving removal of the HI-TRAC from the spent fuel pool shall be sequenced accordingly.

1. Remove the HI-TRAC from the spent fuel pool as follows:
   a. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.
   b. Install the drain line to the underside of the MPC lid. See Figure 8.1.17.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage. See Figure 8.1.11 and 8.1.17.

ALARA Note:
Pre-wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

d. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented. See Figure 8.1.12.

e. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

Note:
The outer diameter of the MPC lid will seat flush with the top edge of the MPC shell when properly installed. Once the MPC lid is installed, the HI-TRAC /MPC removal from the spent fuel pool should proceed in a continuous manner to minimize the rise in MPC water temperature.

f. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

g. Engage the lift yoke to HI-TRAC lifting trunnions.

h. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

ALARA Note:
Activated debris may have settled on the top face of HI-TRAC and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. Users are responsible for any water dilution considerations.

i. Raise HI-TRAC until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Remove any activated or highly radioactive particles from HI-TRAC or MPC.

j. Visually verify that the MPC lid is properly seated. Lower HI-TRAC, reinstall the lid, and repeat as necessary.

k. Install the Lid Retention System bolts if the lid retention system is used.

l. Continue to raise the HI-TRAC under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-TRAC reaches the same elevation as the reservoir, close the Annulus Overpressure System reservoir valve (if used). See Figure 8.1.14.

Caution:

Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action. One course of action involves initiating an MPC water flush for a certain duration and flow rate. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements whenever water is added to the loaded MPC.

m. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

ALARA Note:

Decontamination of HI-TRAC bottom should be performed using remote cleaning methods, covering or other methods to minimize personnel exposure. The bottom lid decontamination may be deferred to a convenient and practical time and location. Any initial decontamination should only be sufficient to preclude spread of contamination within the fuel building.

n. Decontaminate HI-TRAC bottom and HI-TRAC exterior surfaces including the pool lid bottom. Remove the bottom protective cover, if used.

o. If used, disconnect the Annulus Overpressure System from the HI-TRAC See Figure 8.1.14.

p. Set HI-TRAC in the designated cask preparation area.

Note:

If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Users shall evaluate the cask weights to ensure that cask trunnion, lifting devices and equipment load limitations are not exceeded.

q. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.

r. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

Warning:

MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

s. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.

t. Perform decontamination and a dose rate/contamination survey of HI-TRAC.

u. Prepare the MPC annulus for MPC lid welding as follows:

ALARA Note:
If the Temporary Shield Ring is not used, some form of gamma shielding (e.g., lead bricks or blankets) should be placed in the trunnion recess areas of the HI-TRAC water jacket to eliminate the localized hot spot.

v. Decontaminate the area around the HI-TRAC top flange and install the Temporary Shield Ring, (if used). See Figure 8.1.18.

ALARA Note:
The water in the HI-TRAC-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

w. Attach the drain line to the HI-TRAC drain port and lower the annulus water level approximately 6 inches.

2. Prepare for MPC lid welding as follows:

Note:
The following steps use two identical Removable Valve Operating Assemblies (RVOAs) (See Figure 8.1.16) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

Note:
The vent and drain ports are opened by pushing the RVOA handle down to engage the square nut on the cap and turning the handle fully in the counter-clockwise direction. The handle will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the handle fully in the clockwise direction. The ports are closed when the handle cannot be turned further.

Note:
Steps involving preparation for welding may occur in parallel as long as precautions are taken to prevent contamination of the annulus.

a. Clean the vent and drain ports to remove any dirt. Install the RVOAs (See Figure 8.1.16) to the vent and drain ports leaving caps open.

ALARA Warning:
Personnel should remain clear of the drain hoses any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

b. Attach the water pump to the drain port (See Figure 8.1.19) and lower the water level to keep moisture away from the weld region.

c. Disconnect the water pump.

d. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable seal

e. Deflate and remove the inflatable annulus seal.

ALARA Note:
The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontaminated.

f. Survey the MPC lid top surfaces and the accessible areas of the top three inches of the MPC.

ALARA Note:
The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

g. Install the annulus shield. See Figure 8.1.13.

3. Weld the MPC lid as follows:

ALARA Warning:
Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

ALARA Warning:
It may be necessary to rotate or reposition the MPC lid slightly to achieve uniform weld gap and lid alignment. A punch mark is located on the outer edge of the MPC lid and shell. These marks are aligned with the alignment mark on the top edge of the HI-TRAC Transfer Cask (See Figure 8.1.8). If necessary, the MPC lid lift should be performed using a hand operated chain fall to closely control the lift to allow rotation and repositioning by hand. If the chain fall is hung from the crane hook, the crane should be tagged out of service to prevent inadvertent use during this operation. Continuous radiation monitoring is recommended.

a. If necessary center the lid in the MPC shell using a hand-operated chain fall.

Note:
The MPC is equipped with lid shims that serve to close the gap in the joint for MPC lid closure weld.

b. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

ALARA Note:
The AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

c. Install the Automated Welding System baseplate shield. See Figure 8.1.9 for rigging.

d. If used, install the Automated Welding System Robot.

Note:
It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.

Note:
Combustible gas monitoring as described in Step 3e and the associated Caution block are required by the HI-STORM 100 CoC (CoC Appendix B, Section 3.8) and may not be deleted without prior NRC approval via CoC amendment.

Caution:
Oxidation of Boral panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid welding operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space.

e. Perform combustible gas monitoring and exhaust or purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.

f. Perform the MPC lid-to-shell weld and NDE with approved procedures (See 9.1 and Table 2.2.15).

g. Deleted.

h. Deleted.

i. Deleted.

j. Deleted.

4. Perform hydrostatic testing as follows:

ALARA Note:
Testing is performed before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20 for the hydrostatic test arrangement.

ALARA Warning:
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

b. Fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose. Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements.

c. Perform a hydrostatic test of the MPC as follows:

   1. Close the drain valve and pressurize the MPC to 125 +5/-0 psig.

   2. Close the inlet valve and monitor the pressure for a minimum of 10 minutes. The pressure shall not drop during the performance of the test.

   3. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water. The acceptance criteria is no observable water leakage.

d. Release the MPC internal pressure, disconnect the water fill line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.

   1. Repeat the liquid penetrant examination on the MPC lid final pass.

e. Repair any weld defects in accordance with the site's approved weld repair procedures. Reperform the Ultrasonic (if necessary), PT, and Hydrostatic tests if weld repair is performed.

5. Drain the MPC as follows:

a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20.

ALARA Warning:
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

b. Attach the water fill line to the drain port and fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the drain line.

c. Disconnect the water fill and drain lines from the MPC leaving the vent port valve open to allow for thermal expansion of the MPC water.

ALARA Warning:
Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

d. Attach a regulated helium or nitrogen supply to the vent port.

e. Attach a drain line to the drain port shown on Figure 8.1.21.

f. Deleted

g. Verify the correct pressure on the gas supply.

h. Open the gas supply valve and record the time at the start of MPC draining.

Note:
An optional warming pad may be placed under the HI-TRAC Transfer Cask to replace the heat lost during the evaporation process of MPC drying. This may be used at the user's discretion for older and colder fuel assemblies to reduce vacuum drying times.

i. Start the warming pad, if used.

Note:
Users may continue to purge the MPC to remove as much water as possible.

j. Drain the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve. See Figure 8.1.21.

k. Deleted.

l. Disconnect the gas supply line from the MPC.

m. Disconnect the drain line from the MPC.

Note:
Vacuum drying or moisture removal using FHD (for high burn-up fuel) is performed to remove moisture and oxidizing gasses from the MPC. This ensures a suitable environment for long-term storage of spent fuel assemblies and ensures that the MPC pressure remains within design limits. The vacuum drying process described herein reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC. The moisture removal process limits bulk MPC temperatures by continuously circulating gas through the MPC. Section 8.1.5 Steps 6a through o are used for the vacuum drying method of drying and backfill. Section 8.1.5 Steps 7a through i are used for the FHD method of drying and backfill.

6. Dry and Backfill the MPC as follows (Vacuum Drying Method):

   a. Attach the drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a. Other equipment configurations that achieve the same results may also be used.

Note:
The vacuum drying system may be configured with an optional fore-line condenser. Other equipment configurations that achieve the same results may be used.

Note:
To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

b. Open the VDS suction valve and reduce the MPC pressure to below 3 torr.

c. Shut the VDS valves and verify a stable MPC pressure on the vacuum gage.

Note:
The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the vacuum drying system, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

d. Perform the MPC drying pressure test in accordance with the technical specifications.

~~**Caution:**~~
~~Limitations for the at-vacuum duration are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded although a time limit of less than 2 hours at vacuum will bound any MPC.~~

e. Close the vent and drain port valves.

f. Disconnect the VDS from the MPC.

g. Stop the warming pad, if used.

h. Close the drain port RVOA cap and remove the drain port RVOA.

Note:
Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. Other equipment configurations that achieve the same results may be used.

i. Set the helium bottle regulator pressure to the appropriate pressure.

j. Purge the Helium Backfill System to remove oxygen from the lines.

k. Attach the Helium Backfill System to the vent port as shown on Figure 8.1.23 and open the vent port.

l. Slowly open the helium supply valve while monitoring the pressure rise in the MPC.

Note:
If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.

m. Carefully backfill the MPC in accordance with the technical specifications

n. Disconnect the helium backfill system from the MPC.

o. Close the vent port RVOA and disconnect the vent port RVOA.

7. Dry and Backfill the MPC as follows (FHD Method)::

Note:
Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. When using the FHD system to perform the MPC helium backfill, the FHD system shall be evacuated or purged and the system operated with 99.995% (minimum) purity helium.

a. Attach the moisture removal system to the vent and drain port RVOAs. See Figure 8.1.22b. Other equipment configurations that achieve the same results may also be used.

b. Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.

c. Continue the monitoring and moisture removal until LCO 3.1.1 is met for MPC dryness.

d. Continue operation of the FHD system with the demoisturizer on.

e. While monitoring the temperatures into and out of the MPC, adjust the helium pressure in the MPC to provide a fill pressure as required by the technical specifications.

f. Open the FHD bypass line and Close the vent and drain port RVOAs.

g. Close the vent and drain port RVOAs.

h. Shutdown the FHD system and disconnect it from the RVOAs.

i. Remove the vent and drain port RVOAs.

8. Weld the vent and drain port cover plates as follows:

Note:
The process provided herein may be modified to perform actions in parallel.

a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.

b. Place the cover plate over the vent port recess.

c. Weld the cover plate.

Note:
ASME Boiler and Pressure Vessel Code [8.1.3], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

d. Perform NDE on the cover plate with approved procedures (See 9.1 and Table 2.2.15)

e. Repair and weld defects in accordance with the site's approved code weld repair procedures.

f. Deleted.

g. Deleted.

h. Deleted.

i. Repeat for the drain port cover plate.

9. Weld the MPC closure ring as follows:

ALARA Note:
The closure ring is installed by hand. No tools are required. Localized grinding to achieve the desired fit and weld prep are allowed.

a. Install and align the closure ring. See Figure 8.1.8.

b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE with approved procedures (See 9.1 and Table 2.2.15).

c. Deleted.

d. Deleted.

e. Deleted.

f. Deleted.

g. Deleted.

h. Deleted.

i. Deleted.

j. If necessary, remove the AWS. See Figure 8.1.7 for rigging.

8.1.6 Preparation for Storage

ALARA Warning:
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.

Caution:
Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.

1. Remove the annulus shield (if used) and store it in an approved plant storage location
2. If use of the SCS is not required, attach a drain line to the HI-TRAC and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system.
3. Install HI-TRAC top lid as follows:

Warning:
When traversing the MPC with the HI-TRAC top lid using non-single-failure proof (or equivalent safety factors), the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

   a. Install HI-TRAC top lid. Inspect the bolts for general condition. Replace worn or damaged bolts with new bolts.
   b. Install and torque the top lid bolts. See Table 8.1.5 for torque requirements.
   c. Inspect the lift cleat bolts for general condition. Replace worn or damaged bolts with new bolts.
   d. Install the MPC lift cleats and MPC slings. See Figure 8.1.24 and 8.1.25. See Table 8.1.5 for torque requirements.
   e. Drain and remove the Temporary Shield Ring, if used.

4. Replace the pool lid with the transfer lid as follows (Not required for HI-TRAC 125D):

ALARA Note:
The transfer slide is used to perform the bottom lid replacement and eliminate the possibility of directly exposing the bottom of the MPC. The transfer slide consists of the guide rails, rollers, transfer step and carriage. The transfer slide carriage and jacks are powered and operated by remote control. The carriage consists of short-stroke hydraulic jacks that raise the carriage to support the weight of the bottom lid. The transfer step produces a tight level seam between the transfer lid and the pool lid to minimize radiation streaming. The transfer slide jacks do not have sufficient lift capability to support the entire weight of the HI-TRAC. This was selected specifically to limit floor loads. Users should designate a specific area that has sufficient room and support for performing this operation.

Note:
The following steps are performed to pretension the MPC slings.

   a. Lower the lift yoke and attach the MPC slings to the lift yoke. See Figure 8.1.25.
   b. Raise the lift yoke and engage the lift yoke to the HI-TRAC lifting trunnions.

c. If necessary, position the transfer step and transfer lid adjacent to one another on the transfer slide carriage. See Figure 8.1.26. See Figure 8.1.9 for transfer step rigging.

d. Deleted.

e. Position HI-TRAC with the pool lid centered over the transfer step approximately one inch above the transfer step.

f. Raise the transfer slide carriage so the transfer step is supporting the pool lid bottom. Remove the bottom lid bolts and store them temporarily.

ALARA Warning:
Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

g. Lower the transfer carriage and position the transfer lid under HI-TRAC.

h. Raise the transfer slide carriage to place the transfer lid against the HI-TRAC bottom lid bolting flange.

i. Inspect the transfer lid bolts for general condition. Replace worn or damaged bolts with new bolts.

j. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.

k. Raise and remove the HI-TRAC from the transfer slide.

l. Disconnect the MPC slings and store them in an approved plant storage location.

Note:
HI-STORM receipt inspection and preparation may be performed independent of procedural sequence.

5. Perform a HI-STORM receipt inspection and cleanliness inspection in accordance with a site-approved inspection checklist, if required. See Figure 8.1.27 for HI-STORM lid rigging.

Note:
MPC transfer may be performed in the truck bay area, at the ISFSI, or any other location deemed appropriate by the licensee. The following steps describe the general transfer operations (See Figure 8.1.28). The HI-STORM may be positioned on an air pad, roller skid in the cask receiving area or at the ISFSI. The HI-STORM or HI-TRAC may be transferred to the ISFSI using a heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function (See Figure 8.1.29) as long as the HI-TRAC and HI-STORM lifting requirements are not exceeded. The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

8.1.7 <u>Placement of HI-STORM into Storage *(Refer to Section 8.1.1.7 for HI-STORM*</u>

100U)

1. Position an empty HI-STORM module at the designated MPC transfer location. The HI-STORM may be positioned on the ground, on a deenergized air pad, on a roller skid, on a flatbed trailer or other special device designed for such purposes. If necessary, remove the exit vent screens and gamma shield cross plates ,temperature elements and the HI-STORM lid. See Figure 8.1.28 for some of the various MPC transfer options.
    a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.
    b. Transfer the HI-TRAC to the MPC transfer location.
2. De-energize the air pad or chock the vehicle wheels to prevent movement of the HI-STORM during MPC transfer and to maintain level, as required.

ALARA Note:
The HI-STORM vent duct shield inserts eliminate the streaming path created when the MPC is transferred past the exit vent ducts. Vent duct shield inserts are not used with the HI-STORM 100S.

3. Install the alignment device (or mating device for HI-TRAC 125D) and if necessary, install the HI-STORM vent duct shield inserts. See Figure 8.1.30.

Caution:
For MPCs with high burn-up fuel requiring supplemental cooling, the time to complete the transfer may be limited to prevent fuel cladding temperatures in excess of ISG-11 Rev. 3 limits. (See Section 4.5) All preparatory work related to the transfer should be completed prior to terminating the supplemental cooling operations.

4. If used, discontinue the supplemental cooling operations and disconnect the SCS. Drain water from the HI-TRAC annulus to an appropriate plant discharge point.
5. Position HI-TRAC above HI-STORM. See Figure 8.1.28.
6. Align HI-TRAC over HI-STORM (See Figure 8.1.31) and mate the overpacks.
7. If necessary, attach the MPC Downloader. See Figure 8.1.32.
8. Attach the MPC slings to the MPC lift cleats.
9. Raise the MPC slightly to remove the weight of the MPC from the transfer lid doors (or pool lid for HI-TRAC 125D and mating device)
10. If using the HI-TRAC 125D, unbolt the pool lid from the HI-TRAC..
11. Remove the transfer lid door (or mating device drawer) locking pins and open the doors (or drawer).

ALARA Warning:
MPC trim plates are used to eliminate the streaming path above and below the doors (or drawer). If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

12. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.

13. Lower the MPC into HI-STORM.

14. Disconnect the slings from the MPC lifting device and lower them onto the MPC lid.

15. Remove the trim plates (if used), and close the doors (or mating device drawer)

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed due to radiation streaming.
Note:
It may be necessary, due to site-specific circumstances, to move HI-STORM from under the empty HI-TRAC to install the HI-STORM lid, while inside the Part 50 facility. In these cases. users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

16. Remove HI-TRAC from on top of HI-STORM.

17. Remove the MPC lift cleats and MPC slings and install hole plugs in the empty MPC bolt holes. See Table 8.1.5 for torque requirements.

18. Place HI-STORM in storage as follows:

    a. Remove the alignment device (mating device with HI-TRAC pool lid for HI-TRAC 125D)and vent duct shield inserts (if used). See Figure 8.1.30.

    b. Inspect the HI-STORM lid studs and nuts for general condition. Replace worn or damaged components with new ones.

    c. If used, inspect the HI-STORM 100A anchor components for general condition. Replace worn or damaged components with new ones.

    d. Deleted.

Warning:
Unless the lift is single failure proof (or equivalent safety factor) for the HI-STORM Lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

Note:
Shims may be used on the HI-STORM 100 lid studs. If used, the shims shall be positioned to ensure a radial gap of less than 1/8 inch around each stud. The method of cask movement will determine the most effective sequence for vent screen, lid, temperature element, and vent gamma shield cross plate installation.

    e. Install the HI-STORM lid and the lid studs and nuts. See Table 8.1.5 for bolting requirements. Install the HI-STORM 100 lid stud shims if necessary. See Figure 8.1.27 for rigging.

f. Install the HI-STORM exit vent gamma shield cross plates, temperature elements (if used) and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34a and 8.1.34b.

g. Remove the HI-STORM lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location. See Table 8.1.5 for torque requirements.

h. Secure HI-STORM to the transporter device as necessary.

19. Perform a transport route walkdown to ensure that the cask transport conditions are met.

20. Transfer the HI-STORM to its designated storage location at the appropriate pitch. See Figure 8.1.35.

Note:
Any jacking system shall have the provisions to ensure uniform loading of all four jacks during the lifting operation.

a. If air pads were used, insert the HI-STORM lifting jacks and raise HI-STORM. See Figure 8.1.36. Remove the air pad.

b. Lower and remove the HI-STORM lifting jacks, if used.

c. For HI-STORM 100A overpack (anchored), perform the following:

1. Inspect the anchor stud receptacles and verify that they are clean and ready for receipt of the anchor hardware.

2. Align the overpack over the anchor location.

3. Lower the overpack to the ground while adjusting for alignment.

4. Install the anchor connecting hardware (See Table 8.1.5 for torque requirements).

21. Install the HI-STORM inlet vent gamma shield cross plates and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34.

22. Perform shielding effectiveness testing.

23. Perform an air temperature rise test as follows for the first HI-STORM 100 System placed in service:

Note:
The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize. The purpose of this test is to confirm the initial performance of the HI-STORM 100 ventilation system.

a. Measure the inlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average inlet air (or surface screen) temperature.

b. Measure the outlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average outlet air (or surface screen) temperature.

c. Determine the average air temperature rise by subtracting the results of the average inlet screen temperature from the average outlet screen temperature.

d. Report the results to the certificate holder.

Table 8.1.1
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS
125-TON HI-TRAC**

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (Short) overpack (without lid)††	275,000	275,000	275,000					1	
Empty HI-STORM 100S (Tall) overpack (without lid) ††	290,000	290,000	290,000					1	
HI-STORM 100S lid (without rigging. Add 1,000 lbs for 100S Version B Lid)	28,000	28,000	28,000					1	
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	4,200	4,200	4,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Mating Device	15,000	15,000	15,000						
Empty HI-TRAC 125 (without Top Lid, neutron shield jacket water, or bottom lids)	117,803	117,803	117,803	1	1	1			1
Empty HI-TRAC 125D (without Top Lid, neutron shield jacket water, or bottom lids)	122,400	122,400	122,400	1	1	1			1
HI-TRAC 125 Top Lid	2,745	2,745	2,745			1			1
HI-TRAC 125D Top Lid	2,645	2,645	2,645			1			1
Optional HI-TRAC Lid Spacer (weight lbs/in thickness)	400	400	400						
HI-TRAC 125/125D Pool Lid(with bolts)	11,900	11,900	11,900	1	1				
HI-TRAC Transfer Lid (with bolts) (125 Only)	23,437	23,437	23,437			1			1
HI-TRAC 125 Neutron Shield Jacket Water	8,281	8,281	8,281		1	1			1
HI-TRAC 125 D Neutron Shield Jacket Water	9,000	9,000	9,000		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480			1	1		1

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.2 for a description of each load handling case.

†† Short refers to both 100S-232 and 100S Version B-219. Tall refers to both 100S-243 and 100S Version B-229. Weights are based on 200 lb/cf concrete. Add an additional 1955 lbs. for the HI-STORM 100A overpack.

TABLE 8.1.2
ESTIMATED HANDLING WEIGHTS
125-TON HI-TRAC**

Caution:
The maximum weight supported by the 125-Ton HI-TRAC lifting trunnions cannot exceed 250,000 lbs. Users must take actions to ensure that this limit is not exceeded.

Note:
The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine their specific handling weights based on the MPC contents and the expected handling modes.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank empty)	231,700	239,700	238,200
2	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank full)	239,900	248,000	246,500
3	Loaded HI-TRAC 125 During Movement through Hatchway	236,900	244,700	244,100
1A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank empty)	236,400	244,500	243,000
2A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank full)	245,400	253,500	252,000
3A	Loaded HI-TRAC 125D During Movement through Hatchway	230,900	238,700	238,100
4	MPC during transfer operations	80,467	88,315	87,721
5A	Loaded HI-STORM 100 in storage (See Second Note to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (Short) in storage (See Second Note to Table 8.1.1)	380,500	388,600	387,800
5C	Loaded HI-STORM 100S (Tall) in storage (See Second Note to Table 8.1.1)	395,500	403,600	402,800
6	Loaded HI-TRAC and transfer frame during on site handling	239,434	247,282	246,688

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.3
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS100-TON HI-TRAC**

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (Short) overpack (without lid)††	275,000	275,000	275,000					1	
Empty HI-STORM 100S (Tall) overpack (without lid)††	290,000	290,000	290,000					1	
HI-STORM 100S lid (without rigging, add 1,000 lbs for 100S Version B Lid)	28,000	28,000	28,000						
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,200	3,200	3,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Empty HI-TRAC (without Top Lid, neutron shield jacket water, or bottom lids)	84,003	84,003	84,003	1	1	1			1
HI-TRAC Top Lid	1,189	1,189	1,189			1			1
HI-TRAC Pool Lid	7,863	7,863	7,863	1	1				
HI-TRAC Transfer Lid	16,686	16,686	16,686			1			1
HI-TRAC Neutron Shield Jacket Water	7,583	7,583	7,583		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480				1		1

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.4 for a description of each load handling case.

†† Short refers to both 100S-232 and 100S Version B-219. Tall refers to both 100S-243 and 100S Version B-229. Weights are based on 200 lb/cf concrete. Add an additional 1955 lbs. for the HI-STORM 100A overpack.

Table 8.1.4
ESTIMATED HANDLING WEIGHTS
100-TON HI-TRAC**

Caution:
The maximum weight supported by the 100-Ton HI-TRAC lifting trunnions cannot exceed 200,000 lbs. Users must take actions to ensure that this limit is not exceeded.

Note:
The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly and therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine the handling weights based on the contents to be loaded and the expected mode of operations.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC removal from spent fuel pool (neutron tank empty)	192,844	200,942	199,425
2	Loaded HI-TRAC removal from spent fuel pool (neutron tank full)	200,427	208,525	207,008
3	Loaded HI-TRAC During Movement through Hatchway	192,647	200,745	199,901
4	MPC during transfer operations	80,467	88,565	87,721
5A	Loaded HI-STORM 100 in storage (See Second Note to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (Short) in storage (See Second Note to Table 8.1.1)	380,500	388,600	387,800
5C	Loaded HI-STORM 100S (Tall) in storage (See Second Note to Table 8.1.1)	395,500	403,600	402,800
6	Loaded HI-TRAC and transfer frame during on site handling	196,627	204,725	203,881

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.5
HI-STORM 100 SYSTEM TORQUE REQUIREMENTS

Fastener†	Torque (ft-lbs) ††	Pattern†††
HI-TRAC Top Lid Bolts†	Hand tight	None
HI-TRAC Pool Lid Bolts (36 Bolt Lid)†	58 ft-lbs	Figure 8.1.37
HI-TRAC Pool Lid Bolts (16 Bolt Lid)†	110 ft-lbs	Figure 8.1.37
100-Ton HI-TRAC Transfer Lid Bolts†	203 ft-lbs	Figure 8.1.37
125-Ton HI-TRAC Transfer Lid Bolts†	270 ft-lbs	Figure 8.1.37
MPC Lift Cleats Stud Nuts†	793 ft-lbs	None
MPC Lift Hole Plugs†	Hand tight	None
Threaded Fuel Spacers	Hand Tight	None
HI-STORM Lid Nuts †	100 ft-lbs	None
HI-STORM 100S Lid Nuts† (Temporary and Permanent Lids, Including Version B)	Hand Tight	None
Door Locking Pins	Hand Tight + 1/8 to 1/2 turn	None
HI-STORM 100 Vent Screen/Temperature Element Screws	Hand Tight	None
HI-STORM 100A Anchor Studs	55- 65 ksi tension applied by bolt tensioner (no initial torque)	None

† Studs and nuts shall be cleaned and inspected for damage or excessive thread wear (replace if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

†† Unless specifically specified, torques have a +/- 5% tolerance.

††† No detorquing pattern is needed.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure[†]	Description
Air Pads/Rollers	Not Important To Safety	8.1.29	Used for HI-STORM or HI-TRAC cask positioning. May be used in conjunction with the cask transporter or other HI-STORM 100 or HI-TRAC lifting device.
Annulus Overpressure System	Not Important To Safety	8.1.14	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Annulus Shield	Not Important To Safety	8.1.13	A shield that is placed at the top of the HI-TRAC annulus to provide supplemental shielding to the operators performing cask loading and closure operations.
Automated Welding System	Not Important To Safety	8.1.2b	Used for remote field welding of the MPC.
AWS Baseplate Shield	Not Important To Safety	8.1.2b	Provides supplemental shielding to the operators during the cask closure operations.
Bottom Lid Transfer Slide (Not used with HI-TRAC 125D)	Not Important To Safety	8.1.26	Used to simultaneously replace the pool lid with the transfer lid under the suspended HI-TRAC and MPC. Used in conjunction with the bottom lid transfer step.
Cask Transporter	Not Important to Safety unless site-specific conditions require transfer cask or overpack handling outside drop analysis basis.	8.1.29a and 8.1.29b	Used for handling of the HI-STORM 100 Overpack and/or the HI-TRAC Transfer Cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function.
Cool-Down System	Not Important To Safety	8.3.4	A closed-loop forced ventilation cooling system used to gas-cool the MPC fuel assemblies down to a temperature at which water can be introduced without the risk of uncontrolled pressure transients in the MPC due to flashing or thermally shocking the fuel assemblies. The cool-down system is attached between the MPC drain and vent ports. The cool-down system is used only for unloading operations.

[†] Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure[†]	Description
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	8.1.9	Used for rigging such components such as the HI-TRAC top lid, pool lid, MPC lid, transfer lid, AWS, HI-STORM Lid and auxiliary shielding and the empty MPC.
Helium Backfill System	Not Important To Safety	8.1.23	Used for controlled insertion of helium into the MPC for leakage testing, blowdown and placement into storage.
HI-STORM 100 Lifting Jacks	Not Important To Safety	8.1.36	Jack system used for lifting the HI-STORM overpack to provide clearance for inserting or removing a device for transportation. .
Alignment Device	Not Important To Safety	8.1.31	Guides HI-TRAC into place on top of HI-STORM for MPC transfers. (Not used for HI-TRAC 125D)
HI-STORM Lifting Devices	Determined site-specifically based on type, location, and height of lift being performed. Lifting devices shall be provided in accordance with ANSI N14.6.	Not shown.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM 100 for cask handling. Does not include the crane hook (or other primary lifting device) device.
HI-STORM Vent Duct Shield Inserts	Important to Safety Category C .	8.1.30	Used for prevention of radiation streaming from the HI-STORM 100 exit vents during MPC transfers to and from HI-STORM. Not used with the HI-STORM 100S.
HI-TRAC Lid Spacer	Spacer Ring is Not-Important-To-Safety, Studs or bolts are I Important to Safety Category B	Not Shown	Optional ancillary which is used during MPC transfer operations to increase the clearance between the top of the MPC and the underside of the HI-TRAC top lid. Longer threaded studs (or bolts), supplied with the lid spacer, replace the standard threaded studs (or bolts) supplied with the HI-TRAC. The HI-TRAC lid spacer may ONLY be used when the HI-TRAC is handled in the vertical orientation or if HI-TRAC transfer lid is NOT used. The height of the spacer shall be limited to ensure that the weights and C.G. heights in a loaded HI-TRAC with the spacer do not exceed the bounding values found in Section 3.2 of the FSAR.
HI-TRAC Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC handling shall be provided in accordance with ANSI N14.6.	8.1.3	Used for connecting the crane (or other primary lifting device) to the HI-TRAC for cask handling. Does not include the crane hook (or other primary lifting device).

[†] Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure[†]	Description
HI-TRAC transfer frame	Not Important To Safety	8.1.4	A steel frame used to support HI-TRAC during delivery, on-site movement and upending/downending operations.
Cask Primary Lifting Device (Cask Transfer Facility)	Important to Safety. Quality classification of subcomponents determined site-specifically.	8.1.28 and 8.1.32	Optional auxiliary (Non-Part 50) cask lifting device(s) used for cask upending and downending and HI-TRAC raising for positioning on top of HI-STORM to allow MPC transfer. The device may consist of a crane, lifting platform, gantry system or any other suitable device used for such purpose.
Inflatable Annulus Seal	Not Important To Safety	8.1.13	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System	Important to Safety Status determined by each licensee. MPC lid lifting portions of the Lid Retention System shall meet the requirements of ANSI N14.6.	8.1.15, 8.1.17	Optional. The Lid Retention System secures the MPC lid in place during cask handling operations between the pool and decontamination pad.
MPC Lift Cleats	Important To Safety – Category A. MPC Lift Cleats shall be provided in accordance with of ANSI N14.6.	8.1.24	MPC lift cleats consist of the cleats and attachment hardware. The cleats are supplied as solid steel components that contain no weds. The MPC lift cleats are used to secure the MPC inside HI-TRAC during bottom lid replacement and support the MPC during MPC transfer from HI-TRAC into HI-STORM and vice versa. The ITS classification of the lifting device attached to the cleats may be lower than the cleat itself, as determined site-specifically.
Hydrostatic Test System	Not Important to Safety	8.1.20	Used to pressure test the MPC lid-to-shell weld.
MPC Downloader	Important To Safety status determined site-specifically. MPC Downloader Shall meet the requirements of CoC, Appendix B, Section 3.5 .	8.1.28 and 8.1.32	A lifting device used to help raise and lower the MPC during MPC transfer operations to limit the lift force of the MPC against the top lid of HI-TRAC. The MPC downloader may take several forms depending on the location of MPC transfer and may be used in conjunction with other lifting devices.

[†] Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure†	Description
Deleted			
Deleted			
Mating Device	Important-To-Safety – Category B	8.1.31	Used to mate HI-TRAC 125D to HI-ST)RM during transfer operations. Includes sliding drawer for use in removing HI-TRAC pool lid.
MPC Support Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	8.1.25	Used to secure the MPC to the lift yoke during HI-TRAC bottom lid replacement operations. Attaches between the MPC lift cleats and the lift yoke. Can be configured for different crane hook configuration.
MPC Upending Frame	Not Important to Safety	8.1.6	A steel frame used to evenly support the MPC during upending operations. and control the upending process.
Supplemental Cooling System	Important to Safety – Category B	2.C.1	A system used to circulate water or other coolant through the HI-TRAC annulus in order to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits during operations with the MPC in the HI-TRAC. Required only for MPC containing high burn-up fuel as determined in accordance with Section 4.5.
Deleted			
Deleted			
Temporary Shield Ring	Not Important To Safety	8.1.18	A water-filled tank that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.
Vacuum Drying (Moisture Removal) System	Not Important To Safety	8.1.22a	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Not Important To Safety	8.1.22b	Used for removal of residual moisture from the MPC following water draining.
Vent and Drain RVOAs	Not Important To Safety	8.1.16	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Deleted			
Weld Removal System	Not Important To Safety	8.3.2b	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

† Figures are representative and may not depict all configurations for all users.

Table 8.1.7
HI-STORM 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Instrument	Function
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors fluid flow rate during various loading and unloading operations.
Deleted	
Deleted	Ensures leakage rates of welds are within acceptance criteria.
Deleted	
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Deleted	
Temperature Surface Pyrometer	For HI-STORM vent operability testing.
Vacuum Gages	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Deleted	
Deleted	
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

† All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 8.1.8
HI-STORM 100 SYSTEM OVERPACK INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100 overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100 Overpack Lid:

1. Lid studs and nuts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device/ holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.

HI-STORM 100 Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be available, intact, and free of holes and tears in the fabric.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.
7. Anchor hardware, if used, shall be checked for general condition.

Table 8.1.9
MPC INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:

1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Vent and Drain attachments shall be inspected for availability, thread condition operability and general condition.
4. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
5. Lower fuel spacers (if used) shall be inspected for availability and general condition.
6. Drain and vent port cover plates shall be inspected for availability and general condition.
7. Serial numbers shall be inspected for readability.

MPC Main Body:

1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.

Table 8.1.10
HI-TRAC TRANSFER CASK INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-TRAC Top Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All Top Lid surfaces shall be relatively free of dents, scratches, gouges or other damage.

HI-TRAC Main Body:

1. The painted surfaces shall be inspected for corrosion, chipped, cracked or blistered paint.
2. The Top Lid bolt holes shall be inspected for dirt, debris and thread damage.
3. The Top Lid lift holes shall be inspected for thread condition.
4. Lifting trunnions shall be inspected for deformation, cracks, end plate damage, corrosion, excessive galling, damage to the locking plate, presence or availability of locking plate and end plate retention bolts.
5. Pocket trunnion, if used, recesses shall be inspected for indications of overstressing (i.e., cracks, deformation, and excessive wear).
6. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
7. The nameplate shall be inspected for presence and general condition.
8. The neutron shield jacket shall be inspected for leaks.
9. Neutron shield jacket pressure relief valve shall be inspected for presence, and general condition.
10. The neutron shield jacket fill and drain plugs shall be inspected for presence, leaks, and general condition.
11. Bottom lid flange surface shall be clean and free of large scratches and gouges.

Table 8.1.10 (Continued)
HI-TRAC OVERPACK INSPECTION CHECKLIST

HI-TRAC Transfer Lid (Not used with HI-TRAC 125D):

1. The doors shall be inspected for smooth actuation.
2. The threads shall be inspected for general condition.
3. The bolts shall be inspected for indications of overstressing (i.e., cracks, deformation, thread damage, excessive wear) and replaced as necessary.
4. Door locking pins shall be inspected for indications of overstressing (i.e., cracks, and deformation, thread damage, excessive wear) and replaced as necessary.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Lifting holes shall be inspected for thread damage.

HI-TRAC Pool Lid:

1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Threads shall be inspected for indications of damage.

8.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

8.3.1 Overview of HI-STORM 100 System Unloading Operations

> **ALARA Note:**
> The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. Figure 8.3.1 shows a flow diagram of the HI-STORM unloading operations. Figure 8.3.2 illustrates the major HI-STORM unloading operations. *MPC Recovery from HI-STORM 100U is addressed in Section 8.1.3.*

Refer to the boxes of Figure 8.3.2 for the following description. The MPC is recovered from HI-STORM either at the ISFSI or the fuel building using the same methodologies as described in Section 8.1 (Box 1). The HI-STORM lid is removed, the vent duct shield inserts are installed, the alignment device (or mating device with pool lid for HI-TRAC 125D) is positioned, and the MPC lift cleats are attached to the MPC. The exit vent screens and gamma shield cross plates are removed as necessary. MPC slings are attached to the MPC lift cleat and positioned on the MPC lid. HI-TRAC is positioned on top of HI-STORM (Box 2) and the slings are brought through the HI-TRAC top lid. The MPC is raised into HI-TRAC, the HI-TRAC doors (or mating device drawer) are closed and the locking pins are installed. If the mating device and HI-TRAC 125D are used, the pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of HI-STORM. If the HI-TRAC 125D is not used, the HI-TRAC is positioned in the transfer slide and the transfer lid is replaced with the pool lid (Box 3) using the same methodology as with the loading operations.

If the MPC contains high burn-up fuel, a Supplemental Cooling System (SCS) (if required) is connected to the HI-TRAC annulus following transfer from the HI-STORM to the HI-TRAC and used to circulate coolant to maintain fuel cladding temperatures below ISG-11 Rev. 3 limits. HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC slings and MPC lift cleats are removed. The temporary shield ring is installed on the HI-TRAC upper section and filled with plant demineralized water. The HI-TRAC top lid is removed[1] (Box 4) and a water flush is performed on the annulus. Water is fed into the annulus through the drain port and allowed to cool the MPC shell. After a predetermined period (based on the fuel conditions), cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid. The weld removal system is installed (Box 7) and the MPC vent and drain ports are accessed (Box 5). The vent RVOA is attached to the vent port and an

[1] *Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.*

evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is cooled using the cool-down system to reduce the MPC internal temperature to allow water flooding (Box 6). The cool-down process gradually reduces the cladding temperature to a point where the MPC may be flooded with water without thermally shocking the fuel assemblies or over-pressurizing the MPC from the formation of steam. Following the fuel cool-down, the MPC is filled with water (borated as required) and the supplemental cooling is terminated (if used). The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place (Box 7).

The top surfaces of the HI-TRAC and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or lid retention system and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed (Boxes 8 and 9). All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris and crud (Box 10). HI-TRAC and MPC are returned to the designated preparation area (Box 11) where the MPC water is pumped back into the spent fuel pool or liquid radwaste facility. The annulus water is drained and the MPC and overpack are decontaminated (Box 12 and 13).

8.3.2 HI-STORM Recovery from Storage *(Refer to Section 8.1.3 for HI-STORM 100U)*

Note:
The MPC transfer may be performed using the MPC downloader or the overhead crane.

1. Recover the MPC from HI-STORM as follows:
   a. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met.
   b. Transfer HI-STORM to the fuel building or site designated location for the MPC transfer.
   c. Position HI-STORM under the lifting device.
   d. Remove the HI-STORM lid nuts, washers and studs.
   e. Remove the HI-STORM lid lifting hole plugs and install the lid lifting sling. See Figure 8.1.27.

Note:
The specific sequence for vent screen, temperature element, and gamma shield cross plate removal may vary based on the mode(s) or transport.

   f. Remove the HI-STORM exit vent screens, temperature elements and gamma shield cross plates. See Figure 8.1.34a and b.

Warning:
Unless the lift is single-failure proof (or equivalent safety factor) for the HI-STORM lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

g. Remove the HI-STORM lid. See Figure 8.1.27.

h. Install the alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (HI-STORM 100 only). See Figure 8.1.30.

i. Deleted.

j. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings to the MPC lid. See Table 8.1.5 for torque requirements.

k. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for rigging. See Table 8.1.5 for torque requirements.

l. Deleted.

2. If necessary, configure HI-TRAC with the transfer lid (Not required for HI-TRAC 125D):

ALARA Warning:
The bottom lid replacement as described below may only be performed on an empty (i.e., no MPC) HI-TRAC.

a. Position HI-TRAC vertically adjacent to the transfer lid. See Section 8.1.2.

b. Remove the bottom lid bolts and plates and store them temporarily.

c. Raise the empty HI-TRAC and position it on top of the transfer lid.

d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.

e. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.

3. At the site's discretion, perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

Note:
If the HI-TRAC is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

4. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.

5. Engage the lift yoke to the HI-TRAC lifting trunnions.

6. Align HI-TRAC over HI-STORM and mate the overpacks. See Figure 8.1.31.
7. If necessary, install the MPC downloader.
8. Remove the transfer lid (or mating device) locking pins and open the doors (mating device drawer).
9. At the user's discretion, install trim plates to cover the gap above and below the door (drawer for 125D). The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.
10. Attach the ends of the MPC sling to the lifting device or MPC downloader. See Figure 8.1.32.

ALARA Warning:
If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.
Caution:
Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance. For MPCs containing high burn-up fuel, the Supplemental Cooling System (SCS) (if required) is used to prevent fuel cladding temperatures from exceeding ISG-11 Rev. 3 limits. Operation of the SCS typically begins as soon as the MPC is placed in the HI-TRAC and continues until MPC cool-down and re-flooding operations have commenced. Staging and check-out of the SCS shall be completed prior to transferring the MPC to the HI-TRAC to minimize the time required to begin its operation.

11. Raise the MPC into HI-TRAC.
12. Verify the MPC is in the full-up position.
13. Close the HI-TRAC doors (or mating device drawer) and install the door locking pins.
14. For the HI-TRAC 125D, bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.
15. Lower the MPC onto the transfer lid doors (or pool lid for 125D).
16. Disconnect the slings from the MPC lift cleats.

Note:
For the HI-TRAC 100 and HI-TRAC 125, operation of the SCS may need to be postponed until the pool lid is in place on the HI-TRAC. In any event, supplemental cooling shall begin before time limits established by the canister thermal evaluation are exceeded.
Warning:
At the start of SCS operations, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of filling the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased.

17. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1). Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.

18. If necessary, remove the MPC downloader from the top of HI-TRAC.

19. Remove HI-TRAC from the top of HI-STORM.

8.3.3 Preparation for Unloading:

1. Replace the transfer lid with the pool lid as follows (Not required for HI-TRAC 125D):

   a. Lower the lift yoke and attach the MPC slings between the lift cleats and the lift yoke. See Figure 8.1.25.

   b. Engage the lift yoke to the HI-TRAC lifting trunnions.

   c. Deleted.

   d. Raise HI-TRAC and position the transfer lid approximately one inch above the transfer step. See Figure 8.1.26.

   e. Raise the transfer slide carriage so the transfer carriage is supporting the transfer lid bottom. Remove the transfer lid bolts and store them temporarily.

ALARA Warning:
Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

   f. Lower the transfer carriage and position the pool lid under HI-TRAC.

   g. Raise the transfer slide carriage to place the pool lid against the HI-TRAC bottom lid bolting flange.

   h. Inspect the bottom lid bolts for general condition. Replace worn or damaged bolts with new bolts.

   i. Install the pool lid bolts. See Table 8.1.5 for torque requirements.

   j. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1) Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.

   k. Raise and remove the HI-TRAC from the transfer slide.

   l. Disconnect the MPC slings and lift cleats.

   m. Deleted.

   n. Deleted.

2. Place HI-TRAC in the designated preparation area.

Warning:
Unless the lift is single-failure proof (or equivalent safety factor) the HI-TRAC top lid, the top lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

3. Prepare for MPC cool-down as follows:

    a. Remove the top lid bolts and remove HI-TRAC top lid. See Figure 8.1.9 for rigging.

Warning:
At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus flush. Users may also elect the source of water and method for collecting the water flowing from the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased. Water flush should be performed for a minimum of 33 hours at a flow rate of 10 GPM or as specified for the particular heat load of the MPC.. Annulus filling is only required if the SCS is not used.

    b. If necessary, perform annulus flush by injecting water into the HI-TRAC drain port and allowing the water to cool the MPC shell and lid.

4. If necessary, set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid.

5. Access the MPC as follows:

ALARA Note:
The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically vacuumed.

ALARA Warning:
Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

    a. Install bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.

    b. Using the marked locations of the vent and drain ports, core drill the closure ring and vent and drain port cover plates.

6. Remove the closure ring section and the vent and drain port cover plates.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

7. Take an MPC gas sample as follows:

Note:
Users may select alternate methods of obtaining a gas sample.

a. Attach the RVOAs (See Figure 8.1.16).

b. Attach a sample bottle to the vent port RVOA as shown on Figure 8.3.3.

c. Using the vacuum drying system, evacuate the RVOA and Sample Bottle.

d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

g. Deleted.

8. Fill the MPC cavity with water as follows:

a. Configure the cool-down system as shown on Figure 8.3.4.

b. Verify that the helium gas pressure regulator is set to the appropriate pressure.

c. Open the helium gas supply valve to purge the gas lines of air.

d. Deleted.

e. If necessary, slowly open the helium supply valve and increase the Cool-Down System pressure. Close the helium supply valve.

f. Start the gas coolers.

g. Open the vent and drain port caps using the RVOAs.

h. Start the blower and monitor the gas exit temperature. Continue the fuel cool-down operations until the gas exit temperature meets the requirements.

Note:
Water filling should commence immediately at the completion of fuel cool-down operations to prevent fuel assembly heat-up. Prepare the water fill line and the vent line in advance of water filling.

i. Prepare the MPC fill and vent lines as shown on Figure 8.1.20. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Turn off the blower and disconnect the gas lines to the vent and drain port RVOAs. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

Note:
When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.

j. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. (Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

k. If used, cease operation of the SCS and remove the system from the HI-TRAC.

Caution:
Oxidation of Boral panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid cutting operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.

l. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.

m. Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge or evacuate the gas space under the lid as necessary

n. Remove the MPC lid-to-shell weld using the weld removal system. See Figure 8.1.9 for rigging.

o. Vacuum the top surfaces of the MPC and HI-TRAC to remove any metal shavings.

9. Install the inflatable annulus seal as follows:

Caution:
Do not use any sharp tools or instruments to install the inflatable seal.

a. Remove the annulus shield.

b. Manually insert the inflatable seal around the MPC. See Figure 8.1.13.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.

10. Place HI-TRAC in the spent fuel pool as follows:

a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.

b. Engage the lift yoke to HI-TRAC lifting trunnions, remove the MPC lid lifting hole plugs and attach the MPC lid slings or lid retention system to the MPC lid.

c. If the lid retention system is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.

d. Install the lid retention system bolts if the lid retention system is used.

ALARA Note:
The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

e. If used, fill the annulus overpressure system lines and reservoir with demineralized water and close the reservoir valve. Attach the annulus overpressure system to the HI-TRAC. See Figure 8.1.14.

f. Position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

g. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.

h. When the top of the HI-TRAC reaches the elevation of the reservoir, open the annulus overpressure system reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

i. If the lid retention system is used, remove the lid retention bolts when the top of HI-TRAC is accessible from the operating floor.

j. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

k. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.

l. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.

m. Disconnect the drain line from the MPC lid.

n. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.

o. Disconnect the lid retention system if used.

8.3.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Vacuum the cells of the MPC to remove any debris or corrosion products.
3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

8.3.5 Post-Unloading Operations

1. Remove HI-TRAC and the unloaded MPC from the spent fuel pool as follows:

a. Engage the lift yoke to the top trunnions.

b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.

c. Raise HI-TRAC until HI-TRAC flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of HI-TRAC during fuel unloading.

d. Measure the dose rates at the top of HI-TRAC in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

e. Raise the top of HI-TRAC and MPC to the level of the spent fuel pool deck.

f. Close the annulus overpressure system reservoir valve.

g. Using a water pump, lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of HI-TRAC, the surfaces of HI-TRAC and lift yoke should be kept wet until decontamination can begin.

h. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water.

i. Disconnect the annulus overpressure system from the HI-TRAC via the quick disconnect.

j. Place HI-TRAC in the designated preparation area.

k. Disengage the lift yoke.

l. Perform decontamination on HI-TRAC and the lift yoke.

2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

4. Drain the water in the annulus area by connecting the drain line to the HI-TRAC drain connector.

5. Remove the MPC from HI-TRAC and decontaminate the MPC as necessary.

6. Decontaminate HI-TRAC.

7. Remove the bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.

8. Return any HI-STORM 100 equipment to storage as necessary.

SUPPLEMENT 8.I

OPERATING PROCEDURES FOR THE HI-STORM 100U SYSTEM

8.I.0 INTRODUCTION

The operations associated with the use of the HI-STORM 100U System, described in Supplement 1.I, are quite similar to the operations for all other variations of the HI-STORM 100 System. The following sections describe those operations that are, in any respect, unique to the HI-STORM 100U System and thus supplement the information presented in Chapter 8. Where practical, the section number used below directly references the corresponding section in Chapter 8. For example, Subsection 8.I.1.6 supplements or replaces the operations described in Subsection 8.1.6. The guidance provided in this supplement shall be used to develop the site-specific loading procedures for the HI-STORM 100U System, as described in the main body of Chapter 8.

8.I.1 PROCEDURE FOR LOADING THE HI-STORM 100U SYSTEM IN THE SPENT FUEL POOL

8.I.1.1 Overview of Loading Operations

The HI-STORM 100U System differs from the other variations of the HI-STORM 100 System in that the vertical ventilated module (VVM) is an integral part of the ISFSI and cannot be transported on site. The steps required to prepare and load the HI-STORM 100 System up to the point of MPC transfer to the HI-STORM VVM are described in Sections 8.1.2 to 8.1.5 of Chapter 8. For the HI-STORM 100U System, the MPC transfer is performed at the ISFSI after preparation of the MPC for storage operations within the HI-TRAC transfer cask. The loaded HI-TRAC transfer cask may be transported between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or other load handling equipment designed for such applications. The operational steps required to prepare, load the MPC, and transport it to the ISFSI using the HI-TRAC transfer cask are the same for both HI-STORM 100U and the aboveground HI-STORM for the case wherein the MPC transfer for long-term storage occurs at the ISFSI. The detailed operational steps presented in this supplement, therefore, start with the preparation and loading at the ISFSI.

Prior to MPC transfer at the ISFSI, the VVM lid is removed and the empty storage module is inspected. The mating device is positioned on top of the VVM. If used, the Supplemental Cooling System (SCS) is disconnected from the HI-TRAC, the HI-TRAC annulus is drained and the HI-TRAC transfer cask is placed on top of the mating device (Figure 8.I.1). The MPC may be downloaded using the vertical cask crawler, the MPC downloader attached to the transfer cask, or other suitable lifting device. The MPC lifting device is attached to the MPC. The pool lid is removed and the mating device drawer is opened. Optional temporary shielding may be installed, as guided by the licensee's radiation protection program, on or around the mating device. The MPC is lowered into the VVM (Figure 8.I.2). Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the

MPC lid. The temporary shielding is removed, if necessary. The mating device and pool lid may be reattached while on the VVM or may be removed and then reattached . HI-TRAC is removed from on top of the VVM. The MPC lift cleats are removed. Plugs are installed in the empty MPC lifting holes to fill the voids left by the removal of the lift cleat studs. The mating device is removed and the VVM lid is installed. Finally, the temperature monitoring elements and their instrument connections, if used at the ISFSI, are installed, and post-loading performance verification is performed, as required.

8.I.1.2 HI-STORM 100U Receiving Operations

Because the HI-STORM 100U VVM is cast as part of the ISFSI itself, there is no handling of the VVM. For an example of the rigging required to handle the HI-STORM 100U VVM Closure Lid, see Figure 8.I.3.

8.I.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.3.

8.I.1.4 MPC Fuel Loading

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.4.

8.I.1.5 MPC Closure

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.1.5.

8.I.1.6 Preparation for Storage

For the HI-STORM 100U System, these activities are essentially identical to those described in Subsection 8.1.6. However, for the HI-STORM 100U VVM, the receipt and cleanliness inspection should be carried out in accordance with a CoC Holder-approved inspection checklist and Table 8.I.1

8.I.1.7 Placement of MPC/HI-STORM 100U VVM into Storage

Note:
Because the HI-STORM 100U VVM is an integral part of the ISFSI, MPC transfer must take place at the ISFSI.

1. Perform a transport route walkdown to ensure that the cask transport conditions are met.
2. Remove the VVM lid. See Figure 8.I.3 for a lid rigging example and Subsection 3.I.2 for the bounding weight of the lid.
3. Inspect all vent and cavity locations in the VVM cavity for foreign objects. Remove any foreign objects.
4. Install the mating device on the VVM. See Subsection 3.I.2 for the bounding weight of the mating device.
5. Transport the HI-TRAC transfer cask to the ISFSI location using a vertical cask crawler or other suitable transportation device.

Caution:
For MPCs with heat loads requiring supplemental cooling, all preparatory work related to the transfer should be completed prior to terminating supplemental cooling operations to prevent fuel cladding temperatures from exceeding the limits set forth in Chapter 2 of this FSAR.

6. If used, discontinue the supplemental cooling operations and disconnect the SCS. Drain the water from the HI-TRAC annulus to an appropriate plant discharge point.
7. Position the HI-TRAC transfer cask above the VVM.
8. Align HI-TRAC over the VVM and mate the casks using the mating device. See Figure 8.I.1.
9. Attach the MPC slings to the MPC lift cleats and the cask transporter or other suitable downloading device. See Figure 8.I.2.
10. Raise the MPC slightly to remove the weight of the MPC from the HI-TRAC pool lid.
11. Unbolt the pool lid from the HI-TRAC and lower the lid into the mating device.
12. Open the mating device drawer.

ALARA Warning:
Temporary shielding may be used to reduce personnel dose during transfer operations. If ALARA considerations dictate that temporary shielding not be used, personnel must remain clear of the immediate area around the mating device drawer during MPC downloading.

13. At the user's discretion, install temporary shielding to cover the gap above and below the mating device drawer.
14. Lower the MPC into the VVM.
15. Verify that the MPC is fully seated in the VVM
16. Disconnect the MPC slings from the downloading device and lower them onto the MPC lid or remove them from the MPC.
17. Remove the temporary shielding and close the mating device drawer.

Note:
The HI-TRAC pool lid may be reattached while positioned on the VVM or following HI-TRAC and mating device removal from the VVM.

18. Bolt the pool lid back onto the HI-TRAC.

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed to comply with ALARA requirements.

19. Remove the HI-TRAC transfer cask from on top of the VVM.
20. Open the mating device drawer and remove the MPC lift cleats and, if installed, the MPC slings.
21. Install hole plugs in the empty MPC bolt holes.
22. Close the mating device drawer and remove the mating device from on top of the VVM.

Warning:
Unless the lift is single failure proof (or equivalent safety factor) for the VVM lid, the lid shall be preferably kept less than 2 feet above the top surface of the VVM while over the MPC. This lift limit action is purely a defense-in-depth measure: the Closure Lid cannot fall and impact the MPC, as discussed in Supplement 1.I.

23. Install the VVM lid. See Figure 8.I.3 for lid rigging and Subsection 3.I.2 for the bounding lid weight.
24. Remove the VVM lid rigging equipment and install the outlet vent cover.
25. Install the HI-STORM temperature monitoring elements (if used).
26. Perform shielding effectiveness testing, as required.

8.I.2 ISFSI OPERATIONS

ISFSI operations for the HI-STORM 100U System are identical to the activities identified in Section 8.2.

8.I.3 PROCEDURE FOR UNLOADING THE HI-STORM 100U SYSTEM IN THE SPENT FUEL POOL

8.I.3.1 Overview of HI-STORM 100U System Unloading Operations

The MPC is recovered from the HI-STORM 100U VVM at the ISFSI using the same methodologies as described in Subsection 8.I.1, except that the order is basically reversed. The VVM temperature monitoring elements (if used) and lid are removed. The mating device is installed and the mating device drawer is opened. The MPC lift cleats are attached to the MPC. The MPC slings are attached to the MPC lift cleats and positioned on the MPC lid. The mating device drawer is closed and the HI-TRAC is positioned on top of the mating device and VVM.

The pool lid is unbolted from the HI-TRAC and the mating device drawer is opened. The MPC slings are brought through the HI-TRAC top lid and connected to the lift device. The MPC is raised into HI-TRAC and the mating device drawer is closed. The pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of the VVM and transported to the preparation area. The remainder of the unloading operation is carried out in accordance with the operations for the standard HI-STORM 100 System (See Subsection 8.3.3).

8.I.3.2 Preparing HI-STORM 100U System and HI-TRAC for Recovery from Storage

1. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met for transporting the loaded HI-TRAC transfer cask.
2. Perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a written inspection checklist. Transport the HI-TRAC to the ISFSI using the cask transporter or other suitable device.
3. Remove the VVM temperature monitoring equipment (if used).
4. Remove the VVM lid, preferably keeping its height above the top of the CEC Flange to under 2 feet. See Figure 8.I.3 for a rigging example and Subsection 3.I.2 for the bounding lid weight.
5. Install the mating device on the VVM.
6. Open the mating device drawer.
7. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings on the MPC lid. See Table 8.1.5 for torque requirements.
8. Close the mating device drawer.
9. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for example rigging and Table 8.1.5 for torque requirements.
10. If previously drained, fill the neutron shield jacket with plant demineralized water or an approved antifreeze solution as necessary. Ensure that the fill and drain plugs are installed.
11. Align HI-TRAC over the mating device and VVM and mate the casks. See Figure 8.I.2.
12. Unbolt the pool lid from the HI-TRAC and lower into the mating device.
13. Open the mating device drawer.
14. At the user's discretion, install temporary shielding to cover the gap above and below the mating device drawer.
15. Raise the MPC slings up through the HI-TRAC and attach them to the lifting device. See Figure 8.I.1.

ALARA Warning:
If temporary shielding is not used, personnel should remain clear of the immediate drawer area during MPC downloading.
Caution:
Limitations for the handling of the loaded MPC in HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Chaper 4 for requirements on the use of the Supplemental Cooling System (SCS) to maintain fuel cladding temperatures below limits when MPCs with greater than threshold limits are transported in the HI-TRAC. Operation of the SCS normally continues until MPC cool-down and re-flooding operations have commenced. Staging and check-out of the SCS shall be completed prior to transferring the MPC to the HI-TRAC to minimize the time required to begin operations.

16. Raise the MPC into HI-TRAC.
17. Verify the MPC is in the full-up position.
18. Close the mating device drawer.
19. Bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.
20. Lower the MPC onto the pool lid.
21. Disconnect the slings from the lifting device and the MPC lift cleats.

Note:
Operation of the SCS will need to be postponed until the pool lid is in place on the HI-TRAC. Supplemental cooling shall begin, if required, by the provision in Chapter 4.
Warning:
At the start of SCS operations, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of filling the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased.

22. If required, attach the SCS to the HI-TRAC annulus and begin circulating coolant. (See Figure 2.C.1). Continue operation of the SCS until MPC cool-down and re-flooding operations have commenced.
23. Remove HI-TRAC from the top of the VVM.
24. Transport the HI-TRAC to the designated preparation area using the cask transporter or other suitable device.
25. Install the VVM lid to prevent entry of foreign objects into the VVM.

8.I.3.3 Preparation for Unloading

Perform the balance of the unloading operations in accordance with Subsection 8.3.3 beginning at Step 2.

8.I.3.4 MPC Unloading

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.3.4.

8.I.3.5 Post-Unloading Operations

For the HI-STORM 100U System, these activities are identical to those described in Subsection 8.3.5.

8.I.4 MPC TRANSFER TO A HI-STAR 100 OVERPACK FOR TRANSPORT OR STORAGE

For the HI-STORM 100U System, these activities are identical to those described in Section 8.4. MPC transfer to the HI-TRAC from the HI-STORM 100U VVM is addressed in Subsection 8.I.3 above as noted in Section 8.3.

8.I.5 MPC TRANSFER INTO THE HI-STORM 100U VVM DIRECTLY FROM TRANSPORT

For the HI-STORM 100U System, these activities are identical to those described in Section 8.5. MPC transfer to the HI-STORM 100U VVM from the HI-TRAC is addressed in Subsection 8.I.1 above as noted in Section 8.1.

Table 8.I.1
HI-STORM 100U VVM INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100U VVM. Specific findings shall be brought to the attention of the project management for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100U VVM Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
3. The lid shall be inspected for the presence or availability of studs, nuts, and hole plugs.
4. Lid lifting points shall be inspected for dirt, debris, and general condition.
5. Lid lift points shall be inspected for general condition.
6. Vent openings shall be free from obstructions.
7. Vent screens shall be available, intact, and free of holes and tears.
8. Temperature monitoring elements, if used, shall be inspected for availability, function, calibration and provisions for mounting to the VVM outlet air passage.

HI-STORM 100U VVM Main Body:

1. Cooling passages shall be free from obstructions.
2. The interior cavity shall be free of debris, litter, tools, and equipment.
3. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.

FIGURE 8.I.1: HI-TRAC ALIGNMENT AND PLACEMENT ON MATING DEVICE AND HI-STORM 100U* VVM

* The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patents rights laws.

FIGURE 8.I.2: DOWNLOADING MPC INTO HI-STORM 100U* VVM

* The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patents rights laws.

FIGURE 8.I.3: EXAMPLE RIGGING CONFIGURATION FOR THE HI-STORM 100U* VVM LID

* The design features of the HI-STORM 100U System are the exclusive intellectual property of Holtec International under U.S. and international patents rights laws.

SUPPLEMENT 9.I

ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM FOR THE HI-STORM 100U SYSTEM

9.I.0 INTRODUCTION

This supplement addresses fabrication, inspection, test, and maintenance program for the HI-STORM 100U VVM. The HI-STORM 100U System does not require any changes to the acceptance criteria or maintenance requirements for the MPC or the HI-TRAC transfer cask. Therefore, the material on the fabrication, maintenance, testing, and inspection of all components contained in the main body of this chapter remains unchanged and applicable.

As described in Supplement 1.I, the VVM consists of a shop-fabricated CEC (Cavity Enclosure Container) installed below grade and a removable Closure Lid. The CEC is a welded shell-type structure made of low carbon steel plate and bar (or forging) stock. Likewise, the Closure Lid is made of welded and formed steel plates. However, unlike the CEC, the Closure Lid also contains shielding concrete.

By virtue of its underground configuration, the CEC is interfaced by the subgrade along its lateral surface by the top reinforced concrete pad near its flanged upper region and by the support foundation along its bottom surface. The requirements on these interfacing bodies to the extent they are needed to enable the CEC to render its intended function are provided in Supplement 3.I. All requirements pertaining to the manufacturing, inspection, testing, and maintenance of the VVM SSCs are presented in this supplement to comply with the provisions of 10CFR71.24(p).

9.I.I ACCEPTANCE CRITERIA

The design, fabrication, inspection, and testing of the VVM is performed in accordance with the applicable codes and standards specified in Supplement 2.I and on the drawing. Acceptance criteria described in FSAR Section 9.1 for the overpack are also implemented for the HI-STORM 100U VVM, as applicable, and as further particularized below.

9.I.1.1 Manufacturing of VVM Components

The manufacturing of VVM components shall be carried out in accordance with the CoC holder's NRC-approved QA program. All elements of the manufacturing cycle will be established to accord with the Important-to-Safety (ITS) designation of the specific part and the applicable provisions of the referenced codes and standards. The acceptance criteria for the manufactured components apply to each step of the manufacturing evolution, namely (a) supplier selection, (b) preparation of material procurement specifications, (c) preparation of the shop traveler and fabrication procedures, (d) fabrication activities such as forming, bending, plasma cutting, and welding, (e) in-process inspections, (f) final inspection, (g) packaging for shipment, and (h) assembling of the documentation package to serve as the archival evidence of adherence to the quality requirements.

9.I.1.2 Site Construction

Like the above-ground HI-STORM overpacks, the site construction activities on the VVM and its interfacing SSCs (namely, the foundation, the subgrade, and the top reinforced concrete pad) shall be carried out to demonstrate compliance with the technical criteria set forth in this FSAR. The specific requirements, to ensure that the required *critical characteristics*[*] of the VVM and the interfacing SSCs are realized, are summarized below.

a. All site construction processes shall be procedurized, reviewed, and approved in accordance with the CoC holder's NRC-approved QA program.

b. Although the ITS designations apply to VVM components only, all interfacing SSCs shall also be designated as ITS category C to ensure that the construction materials and processes will be subject to the necessary quality assurance regimen to realize the properties specified in Supplement 3.I of this FSAR.

c. Compliance with the requirements in this FSAR shall be demonstrated by appropriate testing and the results documented for archival reference. For example, the strength properties of the subgrade can be established using the classical "plate test".

d. The shielding concrete shall comply with the applicable provisions of Appendix 1.D.

e. [redacted]

f. The corrosion barrier(s) installed on the CEC shall be inspected for absence of damage before the surface becomes inaccessible.

g. The dimensional compliance of the CEC, including its verticality, shall be inspected to establish compliance with the governing construction documents.

h. The installation of the foundation anchors and the corrosion barriers shall be in accordance with written procedures.

i. All VVM surfaces that become inaccessible shall be photographed with sufficient resolution to provide a clear archive of their in-situ state at installation.

The governing processes for manufacturing control shall be prepared to be in complete compliance with this FSAR. For example, the dimensions of the HI-STORM 100U VVM steel components and the properties of the shielding concrete shall be verified to be in accordance

[*] See Glossary for definition.

with FSAR Supplement 2.I, Appendix 1.D (as applicable), and the drawings prior to concrete placement. The dimensional inspection and density measurements shall be documented.

The governing NDE references provided for the above-ground HI-STORM overpack in Table 9.1.4 also apply to the VVM, as do the requirements on the documentation package contents specified in paragraph 9.1.1(13).

In order to receive the Certificate-of-Compliance under the CoC holder's QA program, the manufacturing of the VVM components must meet all of the technical, quality control, procedural (quality assurance) and administrative requirements set forth in the manufacturing program.

Similarly, the low carbon steel plates, bars, and forgings, as applicable, used in the construction of the HI-STORM 100U VVM shall be dimensionally inspected to assure compliance with the requirements on the drawings. Test results shall be documented and become part of the quality documentation package. Dimensional inspections of the Closure Lid and its weight measurement after placement in the shielding concrete shall assure that the required amount of shielding material has been incorporated in the lid.

9.I.1.3 <u>Inspections and Testing</u>

i. Post-Construction Inspection:

Each as-built VVM shall be inspected for final acceptance before it is loaded with fuel. The following inspections define minimum acceptance requirements:

a. [redacted]

b. The Container Flange [redacted] gasket/seal bearing surfaces shall be inspected for their horizontal alignment (within specified tolerance).

c. The Restraint Ring welded to the underside of the Closure Lid shall be checked for fitup with the Container Flange.

d. The inlet and outlet air passages [redacted] shall be inspected for absence of obstruction such as debris and extensive weld spatter.

e. The results of the post-construction inspection shall be incorporated in the VVM's Documentation Package.

ii. Shielding Integrity and Effectiveness Test

Gamma and neutron shielding are provided by the concrete and steel components of the VVM as well by as the surrounding subgrade and surface pad.

As shown in Supplement 5.I, the shielding performance of the HI-STORM 100U system is significantly better than its above-ground counterparts. Further, the long-term shielding effectiveness in the HI-STORM 100U system is also assured to an extremely high level of confidence by virtue of its physical configuration, choice of materials, and design embodiment, as described below:

a. [redacted]

b. [redacted]

c. Aging of foundation, subgrade: Even though a very stiff support foundation is specified, some settlement of the foundation is expected. However, any settlement of the foundation would have no deleterious effect on the extent of shielding to the environment.

Second, while the subgrade is unloaded, except when a transporter is passing over it, some settlement of the subgrade and the above top surface pad is expected. Any subgrade settlement, however, will result in a corresponding compaction of its material, which will improve the subgrade's shielding capability. The settlement of the subgrade will not result in any new loading on the CEC structure (which is autonomously supported on the foundation) or the Closure Lid, which is autonomously supported on the CEC structure. Any visible gap or crevice between the Container Flange and the surface pad should be filled with grout or caulking for both esthetic purposes and enhancement of degradation and corrosion mitigation.

d. Effect of Corrosion: It can be readily deduced from the VVM's design that the only surfaces that are not accessible for corrosion monitoring are the bottom face and outer surface of the CEC and the Foundation Anchor Housing. As discussed in Supplement 3.I, corrosion mitigation measures for these surfaces are prescribed and expected to provide adequate corrosion mitigation for decades beyond the Design Life of the VVM. These corrosion mitigation measures, however, are chiefly required to preserve long-term structural, not shielding, integrity of the CEC because the effect of the CEC shell or Bottom Plate corrosion on the dose to the environment is ostensibly miniscule.

Thus, the potential for reduction in the shielding effectiveness of the HI-STORM 100U system due to corrosion is not a concern.

e. Water Intrusion: Because the underground portion of the HI-STORM 100U VVM contains a thick steel shell structure without any penetrations, intrusion of ground water in the MPC storage cavity is not likely. At some sites, ground water may saturate the surrounding subgrade. A water-saturated subgrade, however, will be more effective (not less) in radiation attenuation and, therefore, seepage of ground water around the HI-STORM 100U vertical space will not threaten loss in its shielding integrity.

f. Materials of Construction: The sole material of construction in the underground portion of the VVM is carbon steel, which is a proven material of long-term shielding endurance under fluence levels that are orders of magnitude greater than those present in the CEC. The Closure Lid is comprised of low carbon steel that encases plain concrete. Concrete, like steel, is proven durable material in a radiation environment that suffers negligible change in its shielding capability over long periods of use. Therefore, material degradation induced reduction in the shielding effectiveness of the VVM is not a credible concern.

In conclusion, long-term shielding effectiveness of the HI-STORM 100U is assured by virtue of its design configuration and its constituent materials.

To benchmark the VVM's shielding capability, operational neutron and gamma shielding effectiveness tests shall be performed as described in Subsection 9.1.5.2 for the aboveground systems.

iii. Thermal Acceptance Test

The thermal acceptance test is performed as described in Subsection 9.1.6 for the aboveground systems.

9.I.1.4 Cask Identification

The MPC and HI-TRAC transfer cask are identified as described in Subsection 9.1.7. See Subsection 9.I.3 for discussion of identification requirements for the VVM.

9.I.1.5 Storm Water Control Test

The VVM is designed to direct storm water and snow/ice melt-off away from the lid where the air passages are located. Any minor amount of moisture that may intrude into the MPC cavity due to wind-driven rain will evaporate in a short period of time due to the movement of heated air in the MPC storage cavity. To verify the effectiveness of the storm water drainage design, a one-time test shall be performed after construction of the first VVM to ensure that the design is effective in directing storm water away from the VVM to the ISFSI's drainage system. The VVM shall be subjected to a water spray that simulates exposure to rainfall of at least 2 inches per hour for at least one hour. At the conclusion of the water spray, the depth of the water (if any) in the bottom of the module cavity shall be measured. Any amount of water accumulation beyond wetting of the Bottom Plate indicates an inadequacy in rain diversion features of the VVM and shall be appropriately corrected. It should be noted that, as stated in Supplement 4.I, any amount of accumulated water in the CEC will not cause any component of the HI-STORM 100U System to exceed its regulatory limit. The only deleterious effect of water is the potential for increased preservative degradation (on the wetted surface of the Bottom Plate, CEC shell and other interfacing components) from prolonged exposure.

9.I.1.6 Inspection Criteria

The inspection and test criteria specified in FSAR Table 9.1.2 shall be implemented for the VVM to the extent they are applicable. No additional inspections or tests other than those specified in this supplement are required. Consistent with Table 9.1.4, no non-destructive examination (NDE) beyond visual examination is required for the VVM steel components.

9.I.2 MAINTENANCE PROGRAM

9.I.2.1 HI-STORM 100U System

The HI-STORM 100U System is totally passive by design and requires minimal normal maintenance to ensure it can perform its design functions. Periodic surveillance (via temperature monitoring or visual or camera-aided inspection of air passages) is required to ensure that the air passages in the lid are not blocked. Preventive or remedial painting of the exposed steel surfaces as part of the user's preventive maintenance program is recommended to mitigate corrosion. Such preventive maintenance activities are typical in scope and complexity to other standard maintenance activities at nuclear power plants. Maintenance activities described in FSAR Section 9.2 and Table 9.2.1 shall be implemented for other HI-STORM 100U System components, namely, the MPC and the HI-TRAC transfer cask, without any change.

The maintenance requirements on the VVM also parallel those for the overground HI-STORM overpack, but are slightly different. In the following, the essentials of the maintenance program for VVM are specified. Because the VVM is a passive structure and is largely protected from the weather due to is underground configuration, the maintenance requirements on it are rather minimal. Nevertheless, a carefully articulated preventive maintenance program is essential for a satisfactory service life.

As is true for all components certified pursuant to this FSAR, the maintenance activities on HI-STORM 100U VVM shall be performed in accordance with a written program that fulfills the requirements of the CoC holder's 10CFR72 Subpart G compliant QA program, the owner site's Safety Plan, and the system's Technical Specification.

Among the QA commitments are performance of maintenance by trained personnel by written procedures, written documentation of the maintenance work performed and of the results obtained. Table 9.I.1 provides a listing of the minimum maintenance activities on the HI-STORM 100U VVM.

Table 9.I.1: Maintenance Activities for the HI-STORM 100U VVM*			
	Activity	Frequency	Comment
1.	CEC cavity is visually inspected	Prior to MPC installation	To ensure that VVM internal components are properly aligned, the surface preservatives on all exposed surfaces are undamaged [redacted] and the cavity is free of visible foreign material.
2.	Lid Examination	Prior to MPC installation	Ensure that the tapped holes for lifting are undamaged, the preservatives on the external surfaces are in good condition and the lid is free of dents and rust stains.
3.	Screen Inspection	Prior to installation of the flanged screen assembly and monthly when in use	Ensure that the screen is undamaged.
4.	Interfacing SSCs	Annually	Ensure that the surface pad is free of visible cracks, the interface between the surface pad and the Container Flange is grouted (or caulked) if necessary, the ISFSI drain system is functional, and ground water system and/or the cathodic protection system (if used) are in working order, and that subgrade settlement counter measures are implemented.
5.	Shielding Effectiveness Test	As required by the Radiation Protection Program	—
6.	ISFSI Settlement	Every five years	Confirm that the VVM settlement is within the range of the Plant's "best estimate". Implement countermeasures if the settlement is determined to be excessive by the CoC holder.

* This table provides maintenance activities for the "100U" VVM that parallel those for HI-STORM overpacks in Table 9.2.1.

Table 9.I.1: Maintenance Activities for the HI-STORM 100U VVM*			
	Activity	**Frequency**	**Comment**
7.	VVM Air Temperature Monitoring System (if used)	Per Licensee's QA Program and manufacturer's recommendations	—
8.	VVM In-Service Inspection	Annually	Ensure that the vent screen assembly fasteners remain coated with preservative, the screen is undamaged, all visible external surfaces are corrosion free, and the air passages are not degraded.
9.	Additional VVM In-Service Inspection for Long-Term Interior and Below-grade Degradation: Includes inspection for foreign material accumulation, corrosion (CEC thinning) and [illegible] degradation	On a site-specific basis per Holtec prepared long-term maintenance guidelines and Licensee's preventive maintenance program. Inspection frequency may increase or decrease depending on the results of previous inspections.	Inspection activities shall be commensurate with site-specific conditions. Under some site conditions, visual inspection of accessible areas is sufficient to determine the general condition of the system. Under site conditions conducive to increased degradation, remotely operated inspection devices, VVM Closure Lid removal or even MPC transfer into a HI-TRAC may be necessary for more extensive inspections. The number of VVMs inspected in a given ISFSI array may vary depending on the licensee's maintenance program and inspections results.

9.I.2.2 Subsystem Maintenance

No new ancillary equipment or sub-system requiring periodic maintenance are required for use of the HI-STORM 100U System.

9.I.2.3 Shielding

As discussed in the foregoing, the radiation shielding capacity of the HI-STORM 100U System is expected to degrade negligibly over time. Therefore, unless the VVM is subjected to an extreme environmental event that imparts stresses or temperatures beyond the design basis limits for the system (i.e., prolonged fire or impact from a beyond-the-design basis large energetic projectile) such that this event could potentially degrade the shielding effectiveness of the VVM, no shielding effectiveness tests beyond that required by the Radiation Protection Program are required over the life of the HI-STORM 100U System.

9.I.3 REGULATORY COMPLIANCE

The information presented in this supplement augments the material in the main body of the chapter to fulfill the regulatory requirements pertaining to the testing and maintenance of the HI-STORM 100U system, resolution of issues concerning adequacy and reliability, and cask identification. Section 9.3 of this FSAR provides the necessary information on all HI-STORM system components except the VVM. This section demonstrates the corresponding compliance information on the VVM.

a. The program for pre-operational testing and initial operations, as required by 10CFR72.24(p) for the VVM, is provided in Section 9.I.1 herein.

b. The maintenance protocol for the VVM, as specified in §72.236(g), is provided in Section 9.I.2 herein.

c. The quality assurance requirements on the design, fabrication, on-site construction of the VVM commensurate with its ITS designation are invoked through Chapter 13 of this FSAR and are summarized in Section 9.I.1 herein, as called for in §72.122(a), §72.122(f), §72.128(a)(1) and §72.24(c).

d. The provisions of §72.82(c) with respect to acceptance criteria and the appropriate test program to ensure compliance with the acceptance criterion are fulfilled by Section 9.I.1 herein.

e. The quality requirements with respect to inspection, testing, and documentation, as set down in §72.212(b)(8), are provided in Section 9.I.1 herein.

f. The VVM, except for its Closure Lid, does not require unique marking other than that required for fuel ISFSI configuration management because it is not a transportable structure. The module is essentially an integral portion of the ISFSI, not a separate cask structure that can be moved by the user. As such, the markings provided on the MPC alone are sufficient to meet the requirements of 10CFR72.236(k). Nevertheless, it is required that each VVM cavity be labeled with an identifier that is unique for the specific site. Further, because the Closure Lid will be subject to handling, consistent with defense-in-depth guidelines for handling heavy loads at nuclear plants, its unique identifier and its bounding weight shall be permanently marked on a readily visible location.

10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM system. This section uses the shielding analysis provided in Chapter 5 and the operations procedures provided in Chapter 8 to develop a dose assessment. The dose assessment is provided in Tables 10.3.1, 10.3.2, and 10.3.3.

The dose rates from the HI-STORM 100 overpack, MPC lid, HI-TRAC transfer cask, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the various loading and unloading operations. The dose rates are also calculated for the various conditions of the cask that may affect the dose rates to the operators (e.g., MPC water level, HI-TRAC annulus water level, neutron shield water level, presence of temporary shielding). The dose rates around the 100-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of ~~46~~*60*,000 MWD/MTU and cooling of 3 years including BPRAs. The dose rates around the 125-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of 75,000 MWD/MTU and cooling of 5 years including BPRAs. The dose rates around the HI-STORM 100 overpack are based on 24 PWR fuel assemblies with a burnup of ~~47,500~~*60,000* MWD/MTU and cooling of 3 years. The selection of these fuel assembly types in all fuel cell locations bound all possible PWR and BWR loading scenarios for the HI-STORM System from a dose-rate perspective. The HI-STORM dose rates used in this chapter were calculated for the HI-STORM 100*S Version B*~~-and 100S~~. ~~This is acceptable because the very conservative burnup and cooling time combination used for the calculations results in dose rates which are representative of the 100S Version B at allowable burnup and cooling time combinations.~~ No assessment is made with respect to background radiation since background radiation can vary significantly by site. In addition, exposures are based on work being performed with the temporary shielding described in Table 10.1.2.

The choice of burnup and cooling times used in this chapter is extremely conservative. The bounding burnup and cooling time that resulted in the highest dose rates around the 100-ton, ~~and~~ 125-ton HI-TRACs, *and HI-STORM 100S Version B* were used.~~-in conjunction with the very conservative burnup and cooling time for the HI-STORM 100 overpack (as discussed in Section 5.1).~~ In addition, including the source term from BPRAs increases the level of conservatism. The maximum dose rate due to BPRAs was used in this analysis. As stated in Chapter 5, using the maximum source for the BPRAs in conjunction with the bounding burnup and cooling time for fuel assemblies is very conservative as it is not expected that burnup and cooling times of the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values. This combined with the already conservative dose rates for the HI-TRACs and HI-STORMs results in an upper bound estimate of the occupational exposure. Users' radiation protection programs will assure appropriate temporary shielding is used based on actual fuel to be loaded and resulting dose rates in the field.

For each step in Tables 10.3.1 through 10.3.3, the operator work location is identified. These correspond to the locations identified in Figure 10.3.1. The relative locations refer to all HI-

STORM Overpacks. The dose rate location points around the transfer cask and overpack were selected to model actual worker locations and cask conditions during the operation. Cask operators typically work at an arms-reach distance from the cask. To account for this, an 18-inch distance was used to estimate the dose rate for the worker. This assessment addresses only the operators that perform work on or immediately adjacent to the cask.

Justification for the duration of operations along with the corresponding procedure Sections from Chapter 8 are also provided in the tables. The assumptions used in developing time durations are based on mockups of the MPC, review of design drawings, walk-downs using other equipment to represent the HI-TRAC transfer cask and HI-STORM 100 overpack the HI-STAR 100 overpack and MPC-68 prototype, consultation with UST&D (weld examination) and consultation with cask operations personnel from Calvert Cliffs Nuclear Power Plant (for items such as lid installation and decontamination). In addition, for the shielding calculations, only the Temporary Shield Ring was assumed to be in place for applicable portions of the operations.

Tables 10.3.1a, 10.3.1b, and 10.3.1c provide a summary of the dose assessment for a HI-STORM 100 System loading operation using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and the 125-ton HI-TRAC 125D respectively. Tables 10.3.2a, 2b, and 2c provide a summary of the dose assessment for HI-STORM 100 System unloading operations using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and 125-ton HI-TRAC 125D respectively. Tables 10.3.3a, 3b, and 3c provide a summary of the dose assessment for transferring the MPC to a HI-STAR 100 overpack as described in Section 8.5 of the operating procedures using the 125-ton HI-TRAC,~~and~~ the 100-ton HI-TRAC, *and 125-ton HI-TRAC 125D*~~transfer cask~~, respectively.

10.3.1 Estimated Exposures for Loading and Unloading Operations

The assumptions used to estimate personnel exposures are conservative by design. The main factors attributed to actual personnel exposures are the age and burnup of the spent fuel assemblies and good ALARA practices. To estimate the dose received by a single worker, it should be understood that a canister-based system requires a diverse range of disciplines to perform all the necessary functions. The high visibility and often critical path nature of fuel movement activities have prompted utilities to load canister systems in a round-the-clock mode in most cases. This results in the exposure being spread out over several shifts of operators and technicians with no single shift receiving a majority of the exposure.

The total person-rem exposure from operation of the HI-STORM 100 System is proportional to the number of systems loaded. A typical utility will load approximately four MPCs per reactor cycle to maintain the current available spent fuel pool capacity. Utilities requiring dry storage of spent fuel assemblies typically have a large inventory of spent fuel assemblies that date back to the reactor's first cycle. The older fuel assemblies will have a significantly lower dose rate than the design basis fuel assemblies due to the extended cooling time (i.e., much greater than the values used to compute the dose rates). Users shall assess the cask loading for their particular fuel types (burnup, cooling time) to satisfy the requirements of 10CFR20 [10.1.1].

For licensees using the 100-Ton HI-TRAC transfer cask, design basis dose rates will be higher (than a corresponding 125-Ton HI-TRAC) due to the decreased mass of shielding. Due to the higher expected dose rates from the 100-Ton HI-TRAC, users may need to use the auxiliary shielding (See Table 10.1.2), and should consider preferential loading, and increased precautions (e.g., additional temporary or auxiliary shielding, remotely operated equipment, additional contamination prevention measures). Actual use of optional dose reduction measures must be decided by each user based on the fuel to be loaded.

10.3.2 Estimated Exposures for Surveillance and Maintenance

Table 10.3.4 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM modules or remote viewing methods instead of performing direct visual observation of the modules. Since security surveillances can be performed from outside the ISFSI, and since the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area, a dose rate of 3 mrem/hour is estimated. Although the HI-STORM 100 System requires only minimal maintenance during storage (e.g. touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Since most of the maintenance is expected to occur outside the actual cask array, a dose rate of 10 mrem/hour is estimated

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @2 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED

† See notes at bottom of Table 10.3.4.

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES[†] (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
Deletted							
Deleted							SIMPLE ATTACHMENT NO TOOLS
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACHM MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @1 PER 2 MINTES
Deleted							
Deleted							
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	117.8	29.5	29.5	100 FT @10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	354.2	106.3	106.3	36 BOLTS @2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
Section 8.1.7							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	*487.4*	*81.2*	*81.2*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*43.9*	*2.9*	*2.9*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*11.7*	*4.9*	*9.8*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*205.5*	*68.5*	*68.5*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*205.5*	*68.5*	*68.5*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*122.7*	*32.7*	*65.4*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*11.7*	*2.0*	*3.9*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*122.7*	*10.2*	*20.5*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASUREMENTS@1/MIN
TOTAL						*2063.1* PERSON-MREM	

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (46,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	3	3.4	6.8	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	3	2.3	4.5	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	3	1.0	2.0	40 FEET @2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	*37.6*	*1.9*	*1.9*	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	*37.6*	*0.3*	*0.3*	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	*90.3*	*9.0*	*18.1*	24 BOLTS @2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	*837.0*	*118.6*	*118.6*	17 FEET @2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	*554.8*	*92.5*	*92.5*	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	*1091.9*	*91.0*	*91.0*	50 SMEARS @10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*282.1*	*2.4*	*2.4*	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	*90.3*	*15.1*	*15.1*	100 FT @10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	*90.3*	*3.0*	*3.0*	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	*1091.9*	*36.4*	*36.4*	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	*69.8*	*7.0*	*14.0*	24 BOLTS @2 BOLT/PERSON MINUTES

† See notes at bottom of Table 10.3.4.

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~46~~60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
MEASURE DOSE RATES AT MPC LID	3	5A	1	*69.8*	*3.5*	*3.5*	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	*1091.9*	*1874.4*	*1874.4*	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	*34.2*	*9.1*	*18.2*	8 SEGMENTS @1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	*34.2*	*14.3*	*14.3*	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	*282.1*	*2.4*	*2.4*	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	*34.2*	*1.1*	*1.1*	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	*34.2*	*1.1*	*1.1*	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	*34.2*	*2.9*	*2.9*	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	*34.2*	*3.4*	*3.4*	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	*34.2*	*1.7*	*1.7*	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	*34.2*	*0.6*	*0.6*	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	*34.2*	*1.1*	*1.1*	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	*34.2*	*11.4*	*34.2*	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	*34.2*	*6.8*	*13.7*	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	*34.2*	*11.4*	*22.8*	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	*34.2*	*4.6*	*9.1*	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	*34.2*	*131.1*	*131.1*	MEASURED DURING WELD MOCKUP TESTING

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	34.2	5.7	5.7	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	34.2	1.1	1.1	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	34.2	25.7	25.7	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	34.2	0.6	0.6	1" THREADED FITTING NO TOOLS
Deleted							
Deleted							
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM () TO VENT AND DRAIN PORT RVOAs	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	79.5	2.7	2.7	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	79.5	2.0	2.0	SINGLE THREADED CONNECTION (1 RVOA)

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	79.5	1.3	1.3	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	79.5	2.0	2.0	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	79.5	2.7	2.7	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	79.5	1.3	1.3	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE VENT AND DRAIN COVER PLATE WELD	100	8A	1	79.5	132.5	132.5	MEASURED DURING WELD MOCKUP TESTING
INSTALL SET SCREWS	2	8A	1	79.5	2.7	2.7	4 SET SCREWS @2/MINUTE
PLUG WELD OVER ET SCREWS	8	8A	1	79.5	10.6	10.6	FOUR SINGLE SPOT WELDS @1 PER 2 MINTES
Deleted							
Deleted							
INSTALL AND ALIGN CLOSURE RING	5	8A	1	79.5	6.6	6.6	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	79.5	245.1	245.1	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	79.5	15.9	15.9	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	79.5	1.3	1.3	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	2190.1	36.5	36.5	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	79.5	13.3	26.5	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	79.5	15.9	15.9	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	363.8	151.6	303.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	79.5	1.3	1.3	8 PLUGS @8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	363.8	24.3	24.3	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	363.8	24.3	48.5	INSTALLED BY HAND NO TOOLS

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	900.9	225.2	225.2	100 FT @10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	2190.1	219.0	219.0	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	2190.1	657.0	657.0	36 BOLTS @2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	363.8	24.3	48.5	INSTALLED BY HAND NO TOOLS
Section 8.1.7							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	900.9	300.3	600.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	35.2	52.8	158.4	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	35.2	1.2	2.3	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	692.0	115.3	230.7	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	692.0	46.1	92.3	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	692.0	46.1	92.3	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	363.8	60.6	121.3	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~46~~*60*,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*122.7*	*32.7*	*65.4*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*11.7*	*2.0*	*3.9*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*122.7*	*10.2*	*20.5*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASUREMENTS@1/MIN
TOTAL						*6628.4* PERSON-MREM	

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1.0	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	2.0	1.5	3.0	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	2.0	0.7	1.3	40 FEET @2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	31.1	1.6	1.6	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	31.1	0.3	0.3	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	46.4	4.6	9.3	24 BOLTS @2/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	142.0	23.7	23.7	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	185.3	15.4	15.4	50 SMEARS @10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	46.4	7.7	7.7	100 FT @10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	46.4	1.5	1.5	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	185.3	6.2	6.2	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	37.3	3.7	7.5	24 BOLTS @2 BOLT/PERSON MINUTES

† See notes at bottom of Table 10.3.4.

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
MEASURE DOSE RATES AT MPC LID	3	5A	1	37.3	1.9	1.9	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	185.3	318.1	318.1	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	18.7	5.0	10.0	8 SEGMENTS @1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	18.7	7.8	7.8	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	18.7	0.6	0.6	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	18.7	0.6	0.6	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	18.7	1.6	1.6	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	18.7	1.9	1.9	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	18.7	0.9	0.9	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	18.7	0.3	0.3	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	18.7	0.6	0.6	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	18.7	6.2	18.7	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	18.7	3.7	7.5	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	18.7	6.2	12.5	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	18.7	2.5	5.0	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE OF LID WELD	230	7A	1	18.7	71.7	71.7	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS

Table 10.3.1c HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	18.7	3.1	3.1	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	18.7	0.6	0.6	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	18.7	14.0	14.0	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	18.7	0.3	0.3	1" THREADED FITTING NO TOOLS
Deleted							
Deleted							
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACHM MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2

Table 10.3.1c HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	37.9	0.9	0.9	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	37.9	1.3	1.3	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	37.9	0.6	0.6	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	37.9	63.2	63.2	MEASURED DURING WELD MOCKUP TESTING
INSTALL SET SCREWS	2	8A	1	37.9	1.3	1.3	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	37.9	5.1	5.1	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
Deleted							
Deleted							
INSTALL AND ALIGN CLOSURE RING	5	8A	1	37.9	3.2	3.2	INSTALLED BY HAND NO TOOLS
PERFORM A NDE ON CLOSURE RING WELDS	185	8A	1	37.9	116.9	116.9	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	37.9	7.6	7.6	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	354.2	5.9	5.9	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	37.9	6.3	12.6	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	37.9	7.6	7.6	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	158.5	66.0	132.1	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	37.9	0.6	0.6	8 PLUGS @8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	158.5	10.6	10.6	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND, NO TOOLS
Section 8.1.7							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
ATTACH MPC LIFT SLINGS TO MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE MATING DEVICE LOCKING PINS AND OPEN DRAWER	40	13B	2	118.5	79.0	158.0	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*487.4*	*81.2*	*81.2*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE MATING DEVICE	10	15A	1	*43.9*	*7.3*	*7.3*	3 BOLTS @2 MINUTES PER BOLT
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*11.7*	*4.9*	*9.8*	INSTALL LID AND HYDRO TORQUE 4 BOLTS

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*205.5*	*68.5*	*68.5*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*205.5*	*68.5*	*68.5*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*122.7*	*32.7*	*65.4*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*11.7*	*2.0*	*3.9*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*122.7*	*10.2*	*20.5*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASUREMENTS@1/MIN
TOTAL							*2017.4* PERSON-MREM

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)							
REMOVE INLET VENT SCREENS	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	*122.7*	*10.2*	*20.5*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	*11.7*	*2.0*	*2.0*	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	*205.5*	*27.4*	*27.4*	4 TEMP. ELEMENTS @2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	*43.9*	*2.9*	*2.9*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS

† See notes at bottom of Table 10.3.4.

Table 10.3.2a HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	118.5	7.9	15.8	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39 6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	117.8	39.3	78.5	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	158.5	10.6	21.1	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	354.2	35.4	35.4	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	354.2	106.3	106.3	36 BOLTS @2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	158.5	26.4	26.4	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @2 FT/MINUTE (CRANE SPEED)

Table 10.3.2a HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL						*924.4* PERSON-MREM	

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport							
REMOVE INLET VENT SCREENS	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	*122.7*	*10.2*	*20.5*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	*11.7*	*2.0*	*2.0*	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	*205.5*	*27.4*	*27.4*	4 TEMP. ELEMENTS @2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	*43.9*	*2.9*	*2.9*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	*692.0*	*115.3*	*230.7*	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	*363.8*	*60.6*	*121.3*	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	*692.0*	*46.1*	*92.3*	INSTALLED BY HAND NO FASTENERS

† See notes at bottom of Table 10.3.4.

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~46~~60,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	*363.8*	*60.6*	*60.6*	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	*692.0*	*46.1*	*92.3*	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	*363.8*	*60.6*	*121.3*	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	*35.2*	*11.7*	*23.5*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	*35.2*	*52.8*	*52.8*	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	*35.2*	*11.7*	*23.5*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	*900.9*	*300.3*	*600.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	*363.8*	*24.3*	*48.5*	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	*2190.1*	*219.0*	*219.0*	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	*2190.1*	*657.0*	*657.0*	36 BOLTS @2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	*363.8*	*60.6*	*60.6*	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	*900.9*	*225.2*	*225.2*	100 FT @10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	*79.5*	*8.0*	*8.0*	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	*34.2*	*1.1*	*1.1*	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	*2190.1*	*18.3*	*18.3*	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	*79.5*	*11.9*	*11.9*	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	*34.2*	*22.8*	*45.6*	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES[†] (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	*79.5*	*1.3*	*1.3*	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	*79.5*	*2.7*	*2.7*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	*79.5*	*0.7*	*0.7*	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	*79.5*	*0.7*	*0.7*	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	*79.5*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	*79.5*	*2.7*	*2.7*	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	*79.5*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	*34.2*	*5.7*	*5.7*	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	*79.5*	*1.3*	*1.3*	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	*34.2*	*5.7*	*11.4*	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*282.1*	*9.4*	*9.4*	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*282.1*	*9.4*	*9.4*	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	*69.8*	*2.3*	*2.3*	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	*69.8*	*14.0*	*14.0*	24 BOLTS @2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*282.1*	*2.4*	*2.4*	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	*282.1*	*47.0*	*47.0*	100 FT @10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	*837.0*	*118.6*	*118.6*	17 FEET @2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	*90.3*	*18.1*	*18.1*	24 BOLTS @2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	3	1.0	2.0	40 FEET @2 FT/MINUTE (CRANE SPEED)

Table 10.3.2b HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~46~~*60*,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)							
ACTION	**DURATION (MINUTES)**	**OPERATOR LOCATION (FIGURE 10.3.1)**	**NUMBER OF OPERATORS**	**DOSE RATE AT OPERATOR LOCATION (MREM/HR)**	**DOSE TO INDIVIDUAL (MREM)**	**TOTAL DOSE (PERSON-MREM)**	**ASSUMPTIONS**
REMOVE MPC LID	20	2	2	3	1.0	2.0	CONSULTATION WITH CALVERT CLIFFS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL						*3275.0* PERSON-MREM	

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)							
REMOVE INLET VENT SCREENS	20	16D	1	122.7	40.9	40.9	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	122.7	10.2	20.5	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	122.7	8.2	8.2	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	69.7	46.5	46.5	200 FEET @4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	11.7	2.0	2.0	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	205.5	27.4	27.4	4 TEMP. ELEMENTS @2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
INSTALL MATING DEVICE WITH POOL LID	10	15A	1	43.9	7.3	7.3	3 BOLTS AT 2 MINUTES PER BOLT
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED

† See notes at bottom of Table 10.3.4.

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	158.5	26.4	26.4	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE MATING DEVICE DRAWER AND BOLT-UP POOL LID	36	13B	2	118.5	71.1	142.2	2 PINS@2MIN/PIN, 16 BOLTS @2MIN/BOLT
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	158.5	26.4	52.8	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	26.4	39.6	39.6	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	26.4	8.8	17.6	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	117.8	29.5	29.5	100 FT @10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	37.9	3.8	3.8	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	18.7	0.6	0.6	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	354.2	3.0	3.0	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	37.9	5.7	5.7	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	18.7	12.5	24.9	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	37.9	0.6	0.6	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	37.9	1.3	1.3	SINGLE THREADED CONNECTION (1 RVOA)

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	37.9	0.3	0.3	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	37.9	0.3	0.3	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	37.9	1.3	1.3	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	37.9	0.6	0.6	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	18.7	3.1	3.1	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	37.9	0.6	0.6	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	18.7	3.1	6.2	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	82.7	2.8	2.8	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	37.3	1.2	1.2	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	37.3	7.5	7.5	24 BOLTS @2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	82.7	0.7	0.7	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	82.7	13.8	13.8	100 FT @10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	117.8	16.7	16.7	17 FEET @2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	46.4	9.3	9.3	24 BOLTS @2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	2.0	0.7	1.3	40 FEET @2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	2.0	0.7	1.3	CONSULTATION WITH CALVERT CLIFFS

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL							787.5 PERSON-MREM

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	7.1	1.2	2.4	ALIGN AND POSITION REMOVE 4 SHACKLES
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING

† See notes at bottom of Table 10.3.4.

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	487.4	203.1	406.2	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	118.5	7.9	15.8	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	487.4	81.2	81.2	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	43.9	1.5	1.5	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	43.9	2.9	2.9	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	11.7	4.9	9.8	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	205.5	13.7	13.7	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	205.5	68.5	68.5	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	205.5	68.5	68.5	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	11.7	0.4	0.4	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	11.7	0.4	0.4	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	122.7	32.7	32.7	16POINTS@1 MIN

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*11.7*	*2.0*	*2.0*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	*122.7*	*10.2*	*10.2*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASMT@1/MIN
TOTAL						*1068.3* PERSON-MREM	

Table 10.3.3b
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	6.9	1.8	1.8	8 SEGMENTS @2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	6.9	2.9	2.9	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	6.9	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	6.9	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	6.9	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	6.9	4.5	9.0	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	6.9	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	6.9	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	6.9	1.2	2.3	ALIGN AND POSITION REMOVE 4 SHACKLES
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	487.4	16.2	16.2	4 PLUGS AT 2/MIN NO TORQUING

† See notes at bottom of Table 10.3.4.

Table 10.3.3b MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (4660,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	*487.4*	*203.1*	*406.2*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	*692.0*	*115.3*	*230.7*	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	*692.0*	*46.1*	*92.3*	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	*692.0*	*46.1*	*92.3*	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	*692.0*	*46.1*	*92.3*	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	*363.8*	*60.6*	*121.3*	2 SLINGS@5/MIN
REMOVE TRIM PLATES	4	13B	2	*692.0*	*46.1*	*92.3*	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*487.4*	*81.2*	*81.2*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*43.9*	*2.9*	*2.9*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*11.7*	*4.9*	*9.8*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*205.5*	*68.5*	*68.5*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*205.5*	*68.5*	*68.5*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	*122.7*	*32.7*	*32.7*	16POINTS@1 MIN

Table 10.3.3b
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (4~~6~~*60*,000 MWD/MTU, 3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*11.7*	*2.0*	*2.0*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	*122.7*	*10.2*	*10.2*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASMT@1/MIN
TOTAL						*1633.3* PERSON-MREM	

Table 10.3.3c MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	6.7	0.2	0.2	4 SHACKLES@2/MIN

† See notes at bottom of Table 10.3.4.

Table 10.3.3c MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL MATING DEVICE ON HI-STAR	20	19B	2	7.1	2.4	4.7	ALIGN AND BOLT INTO PLACE
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	*487.4*	*203.1*	*406.2*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	118.5	19.8	39.5	ALIGNMENT GUIDES USED
REMOVE LOCKING PINS AND OPEN DRAWER	4	20B	2	118.5	7.9	15.8	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
RAISE THE POOL LID AND BOLT INTO PLACE ON HI-TRAC	32	20B	2	118.5	63.2	126.4	2 MINS/BOLT, 16 BOLTS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	158.5	26.4	52.8	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	118.5	7.9	15.8	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*487.4*	*81.2*	*81.2*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*487.4*	*16.2*	*16.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*43.9*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE THE MATINGE DEVICE	6	15A	1	*43.9*	*4.4*	*4.4*	3 BOLTS AT 2 MINUTES PER BOLTS
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*11.7*	*4.9*	*9.8*	INSTALL LID AND HYDRO TORQUE 4 BOLTS

Table 10.3.3c MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES[†] (75,000 MWD/MTU, 5-YEAR COOLED PWR FUEL)							
ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*205.5*	*13.7*	*13.7*	4 PCS @1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*205.5*	*68.5*	*68.5*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*205.5*	*68.5*	*68.5*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*11.7*	*0.4*	*0.4*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*11.7*	*0.4*	*0.4*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	*122.7*	*32.7*	*32.7*	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*11.7*	*2.0*	*2.0*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*69.7*	*46.5*	*46.5*	200 FEET @4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS @1/MIN
REMOVE AIR PAD	5	16D	1	*122.7*	*10.2*	*10.2*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*122.7*	*8.2*	*8.2*	4 JACKS @1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*122.7*	*40.9*	*40.9*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*205.5*	*27.4*	*27.4*	8 MEASMT@1/MIN
TOTAL							*1198.6* PERSON-MREM

Table 10.3.4
ESTIMATED EXPOSURES FOR HI-STORM 100 SURVEILLANCE AND MAINTENANCE

ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 10.3.1a, 10.3.1b, 10.3.1c, 10.3.2a, 10.3.2b, 10.3.2c, 10.3.3a, 10.3.3b, 10.3.3c and 10..3.4:

1. Refer to Chapter 8 for detailed description of activities.
2 Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.
3 HI-STAR 100 Operations assume that the cooling time is at least 10 years.

10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT

10.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [10.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STORM 100 System with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10CFR72.212 [10.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.9 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [10.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. As a summary of Chapter 5, Table 10.4.1 presents the annual dose results for a single overpack at 100 and ~~250~~*350* meters and a 2x5 array of HI-STORM 100 systems at ~~450~~*550* meters. These annual doses are based on a full array of design basis fuel with a burnup of ~~47,500~~*60,000* MWD/MTU and 3-year cooling. This burnup and cooling time combination conservatively bounds the allowable burnup and cooling times listed in Section 2.1.9. In addition, 100% occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, the casks were positioned on an infinite slab of soil to account for earth-shine effects. These results indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of ~~250~~*350* meters for a single cask and at ~~450~~*550* meters for a 2x5 array of HI-STORM 100 Systems containing design basis fuel. These results are presented only as an illustration to demonstrate that the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1]. Neither the distances nor the array configurations become part of the Technical Specifications. Rather, users are required to perform a site specific analyses to demonstrate compliance with 10CFR72.104[10.0.1] contributors and 10CFR20[10.1.1].

An additional contributor to the controlled area boundary dose is the loaded HI-TRAC transfer cask, if the HI-TRAC is to be used at the ISFSI outside of the fuel building.

TRAC is used outside at the ISFSI, the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1] when worst-case design basis fuel is loaded in all fuel cell locations. However, users are required to perform a site specific analysis to demonstrate compliance with 10CFR72.104[10.0.1] and 10CFR20[10.1.1] taking into account the actual site boundary distance and fuel characteristics.

Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. 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 ~~that~~ the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 10.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

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

The worst-case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-STORM overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded.

However, with over 25 inches of concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-TRAC transfer cask assumes that as a result of a fire, tornado missile, or handling accident, the all the water in the water jacket is lost. The shielding analysis of the 100-ton HI-TRAC transfer cask with complete loss of the water from the water jacket is discussed in Section 5.1.2. These results bound those for the 125-Ton HI-TRAC transfer cask by a large margin. The results in that section show that the resultant dose rate at the 100-meter controlled area boundary would be approximately 4.~~28~~*22* mrem/hour for the loaded HI-TRAC transfer cask during the accident condition. At the calculated dose rate, it would take approximately ~~48~~*49* days for the dose at the controlled area boundary to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 11. Therefore, the dose requirement of 10CFR72.106 [10.0.1] is satisfied. Once again, this dose is calculated assuming design basis fuel in all fuel cell locations. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 10.4.1

ANNUAL DOSE FOR ARRAYS OF HI-STORM 100*S VERSION B* OVERPACKS
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
~~47,500~~*60,000* MWD/MTU AND 3-YEAR COOLING

Array Configuration	1 Cask	1 Cask	2x5 Array
Annual Dose (mrem/year)†	*883.27*	*19.26*	*16.34*
Distance to Controlled Area Boundary (meters)††, †††	100	~~250~~*350*	~~450~~*550*

† 100% occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 3-year cooling as specified in the Section 2.1.9 is lower than the burnup analyzed for the design basis fuel used in this table.

Table 10.4.2
DOSE RATE FOR THE 100-TON HI-TRAC TRANSFER CASK
WITH DESIGN BASIS ZIRCALOY CLAD FUEL

Fuel Burnup & Cooling Time	**100 Meters**	**200 Meters**	**300 Meters**
~~46~~*60*,000 MWD/MTU & 3 Years	~~0.98~~*1.19* mrem/hr	~~0.15~~*0.18* mrem/hr	~~0.04~~*0.05* mrem/hr
~~75,000 MWD/MTU & 5 Years~~	~~0.80 mrem/hr~~	~~0.12 mrem/hr~~	~~0.03 mrem/hr~~

SUPPLEMENT 10.I

RADIATION PROTECTION

The HI-STORM 100U is a modular, underground vertical ventilated module (VVM) designed to accept all MPC models for storage at an ISFSI in lieu of above ground overpacks, like HI-STORM 100 and HI-STORM 100S. As such, the radiological dose to plant personnel as well as members of the general public is well below those of the HI-STORM 100 and HI-STORM 100S when the MPC is in the overpack. Since the determination of off-site doses is necessarily site-specific, dose assessments similar to those described in Chapter 10 are to be prepared by the licensee as part of implementing the HI-STORM 100U System in accordance with 10CFR72.212 [10.0.1].

The operations associated with the use of HI-STORM 100U, described in Supplements 1.I and 8.I, are quite similar to the operations for all other variations of the HI-STORM 100 system. Therefore, the operator dose rates will be the same, and in some cases lower than, the dose rates described in Chapter 10. Occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections may be calculated using the information presented in the tables of Chapter 10 for the site-specific application of the HI-STORM 100U system. For the fuel loading/unloading, transportation, and storage operations utilizing HI-STORM 100U, the dose information provided in Chapter 10 may be considered bounding.

CHAPTER 11[†]: ACCIDENT ANALYSIS

This chapter presents the evaluation of the HI-STORM 100 System for the effects of off-normal and postulated accident conditions. The design basis off-normal and postulated accident events, including those resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Subsections 2.2.2 and 2.2.3. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, confinement, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM 100 System are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and evaluations described therein.

Chapter 11 is in full compliance with NUREG-1536; no exceptions are taken.

11.1 <u>OFF-NORMAL CONDITIONS</u>

Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions, which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are listed in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM 100:

Off-Normal Pressure
Off-Normal Environmental Temperature
Leakage of One MPC Seal Weld
Partial Blockage of Air Inlets
Off-Normal Handling of HI-TRAC Transfer Cask
Malfunction of FHD System
SCS Power Failure
Off-Normal Loads[‡]

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† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1). This chapter has been substantially revised in support of LAR#3 to enhance clarity of presentation and evaluation of results. Because of extensive editing a clean chapter is issued with this amendment.

‡ Off-normal load combinations are defined in Chapter 2, Table 2.2.14 and evaluated in Chapter 3, Section 3.4.

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events without affecting function, and are in compliance with the applicable acceptance criteria. The following subsections present the evaluation of the HI-STORM 100 System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.104(a) and 10CFR20.

11.1.1 Off-Normal Pressures

The sole pressure boundary in the HI-STORM 100 System is the MPC enclosure vessel. The off-normal pressure condition is specified in Subsection 2.2.2. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the MPC cavity under normal storage. The MPC internal pressure is evaluated with 10% of the fuel rods ruptured and 100% of the rods fill gas and 30% of the fission gases released to the cavity.

11.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is low. Nonetheless, the event is postulated and evaluated.

11.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure and, therefore, no monitoring is required.

11.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off normal internal pressure is reported in Subsection 4.6.1 for the following conditions: limiting fuel storage scenario, tech. spec. maximum helium backfill and 10% rod rupture with 100% of rod fill gas and 30% of gaseous fission products released into the MPC cavity. The analysis shows that the MPC pressure remains below the design MPC internal pressure (Table 2.2.1).

It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would decrease the temperature rise, the calculated pressure is higher than would actually occur.

Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

Thermal

The MPC internal pressure for off-normal conditions is reported in Subsection 4.6.1.. The design basis internal pressure for off-normal conditions used in the structural evaluation (Table 2.2.1) bounds the off-normal condition pressure..

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100 System.

11.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There is no corrective action requirement for off-normal pressure.

11.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.1.2 Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (HI-STORM overpack) and 0 to 100°F (HI-TRAC transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM 100 System and must be evaluated against the allowable component design temperatures. The off-normal temperatures are evaluated against the off-normal condition temperature limits listed in Table 2.2.3.

11.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

11.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM overpack and MPC. Chapter 2 provides operational limitations to the use of the HI-TRAC transfer cask at temperatures of ≤32°F and prohibits use of the HI-TRAC transfer cask below 0°F.

11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considers an environmental temperature of 100°F with insolation for a duration sufficient to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The Off-Normal ambient temperature condition is evaluated in Subsection 4.6.1. The results are in compliance with off-normal temperature and pressure limits in Tables 2.2.1 and 2.2.3.

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM overpack. The HI-STORM overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is 0°F and the HI-TRAC is conservatively assumed to reach 0°F throughout the structure. For ambient temperatures from 0° to 32°F, antifreeze must be added to the demineralized water in the water jacket to prevent freezing. Chapter 3, Subsection 3.1.2, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100 System.

Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 4.6.1, the resultant pressure is below the off-normal design pressure (Table 2.2.1). The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsection 3.1.2.

Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Subsection 4.6.1 for the HI-STORM overpack and MPC. As can be seen from this table, all temperatures for the off-normal environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100 System.

11.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no corrective actions required for off-normal environmental temperatures.

11.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

11.1.3 Leakage of One Seal

The HI-STORM 100 System has a reliable welded boundary to contain radioactive fission products within the confinement boundary. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, vent and drain port cover plates, and associated welds. The closure ring provides a redundant welded closure to the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. Additionally, the MPC lid weld is subjected to a pressure test to verify its integrity.

Section 7.1 provides a discussion as to how the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.

11.1.3.1 Postulated Cause of Leakage of One Seal in the Confinement Boundary

There is no credible cause for the leakage of one seal in the confinement boundary. The conditions analyzed in Chapter 3 shows that the confinement boundary components are maintained within their Code-allowable stress limits under all normal and off-normal storage conditions. The MPC fabrication and closure welds meet the requirements of ISG-18, such that leakage from the confinement boundary is not considered credible. Therefore, there is no event that could cause leakage of one seal in the confinement boundary.

11.1.3.2 Detection of Leakage of One Seal in the Confinement Boundary

The HI-STORM 100 System is designed such that leakage of one field weld in the confinement boundary is not considered a credible scenario. Therefore, there is no requirement to detect leakage from one seal.

11.1.3.3 Corrective Action for Leakage of One Seal in the Confinement Boundary

There is no corrective action required for the failure of one weld in the closure system of the confinement boundary. Leakage of one weld in the confinement boundary closure system is not a credible event.

11.1.3.4 Radiological Impact of Leakage of One Seal in the Confinement Boundary

The off-normal event of the failure of one weld in the confinement boundary closure system has no radiological impact because leakage from the confinement barrier is not credible.

11.1.4 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with debris screens on the inlet and outlet air ducts. These screens ensure the air ducts are protected from the incursion of foreign objects. There are four air inlet ducts 90° apart and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet ducts. As required by the design criteria presented in Chapter 2, it is conservatively assumed that two of the four air inlet ducts are blocked. The blocked air inlet ducts are assumed to be completely blocked with an ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the inherent thermal stability of the HI-STORM 100 System.

11.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is conservatively assumed that the blocked air inlet ducts are completely blocked, although screens prevent foreign objects from entering the ducts. The screens are either inspected periodically or the outlet duct air temperature is monitored. It is, however, possible that blowing debris may block two air inlet ducts of the overpack.

11.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet ducts will occur during the routine visual inspection of the screens or temperature monitoring of the outlet duct air. The frequency of inspection is based on an assumed complete blockage of all four air inlet ducts. There is no inspection requirement as a result of the postulated two inlet duct blockage, because the complete blockage of all four air inlet ducts is bounding.

11.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

The thermal analysis for the two air inlet ducts blocked off-normal condition is performed in Subsection 4.6.1. The analysis demonstrates that under bounding (steady-state) conditions, no system components exceed the off-normal temperature limits in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100 System.

11.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet ducts does not affect the HI-STORM 100 System's ability to operate safely.

Periodic inspection of the HI-STORM overpack air duct screens is required. Alternatively, the outlet duct air temperature is monitored. The frequency of inspection is based on an assumed blockage of all four air inlet ducts analyzed in Section 11.2.

11.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet ducts has no radiological impact because the confinement barrier is not breached and shielding is not affected.

11.1.5 Off-Normal Handling of HI-TRAC

During upending and/or downending of the HI-TRAC transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation.

If the lifting device is allowed to "go slack", the total weight would be applied to the lower pocket trunnions only. Under this off-normal condition, the pocket trunnions would each be required to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to demonstrate that the pocket trunnions possess sufficient strength to support the increased load under this off-normal condition.

This off-normal condition does not apply to the HI-TRAC 125D, which does not have lower pocket trunnions. Upending and downending of the HI-TRAC 125D is performed using an L-frame.

11.1.5.1 Postulated Cause of Off-Normal Handling of HI-TRAC

If the cable of the crane handling the HI-TRAC is inclined from the vertical, it would possible to unload the upper lifting trunnions such that the lower pocket trunnions are supporting the total cask weight and the lifting trunnions are only preventing cask rotation.

11.1.5.2 Detection of Off-Normal Handling of HI-TRAC

Handling procedures and standard rigging practice call for maintaining the crane cable in a vertical position by keeping the crane trolley centered over the lifting trunnions. In such an orientation it is not possible to completely unload the lifting trunnions without inducing rotation. If the crane cable were inclined from the vertical, however, the possibility of unloading the lifting trunnions would exist. It is therefore possible to detect the potential for this off-normal condition by monitoring the incline of the crane cable with respect to the vertical.

11.1.5.3 Analysis of Effects and Consequences of Off-Normal Handling of HI-TRAC

If the upper lifting trunnions are unloaded, the lower pocket trunnions will support the total weight of the loaded HI-TRAC. The analysis of the pocket trunnions to support the applied load of one-half of the total weight is provided in Subsection 3.4.4. The consequence of off-normal handling of the HI-TRAC is that the pocket trunnions safely support the applied load.

Structural

The stress evaluations of the lower pocket trunnions are discussed in Subsection 3.4.4. All stresses are within the allowable values.

Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal handling of the HI-TRAC does not affect the safe operation of the system.

11.1.5.4 Corrective Action for Off-Normal Handling of HI-TRAC

The HI-TRAC transfer casks are designed to withstand the off-normal handling condition without any adverse effects. There are no corrective actions required for off-normal handling of HI-TRAC other than to attempt to maintain the crane cable vertical during HI-TRAC upending or downending.

11.1.5.5 Radiological Consequences of Off-Normal Handling of HI-TRAC

The off-normal event of off-normal handling of HI-TRAC has no radiological impact because the confinement barrier is not breached and shielding is not affected.

11.1.6 Malfunction of FHD System

The FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, the FHD system is equipped with active components requiring external power for normal operation.

11.1.6.1 Postulated Cause of FHD Malfunction

Likely causes of FHD malfunction are (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in any helium leakage from the MPC or the FHD itself.

11.1.6.2 Detection of FHD Malfunction

The FHD System is monitored during its operation. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

11.1.6.3 Analysis of Effects and Consequences of FHD Malfunction

Structural

The FHD System is required to be equipped with safety relief devices§ to prevent the MPC structural boundary pressures from exceeding the design limits. Consequently there is no adverse effect.

Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit is 1058°F (Table 2.2.3). The FHD System malfunction event is evaluated assuming the following bounding conditions:

1) Steady state maximum temperatures have been reached
2) Design basis heat load
3) Standing column of air in the annulus
4) MPCs backfilled with the minimum helium pressure required by the Technical Specifications

It is noted that operator action may be required to raise the helium regulator set point to ensure that condition 4 above is satisfied. These conditions are the same as for the normal on-site transfer in a vertically oriented HI-TRAC, discussed in Section 4.5. The steady state results are provided in Table 4.5.4. The results demonstrate that the peak fuel cladding temperatures remain below the limit in the event of a prolonged unavailability of the FHD system.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, the structural boundary pressures cannot exceed the design limits.

§ The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.

Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the FHD malfunction does not affect the safe operation of the HI-STORM 100 System.

11.1.6.4 Corrective Action for FHD Malfunction

The HI-STORM 100 System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently no corrective action is required.

11.1.6.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.1.7 SCS Power Failure

The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation.

11.1.7.1 Postulated Cause of SCS Power Failure

The SCS is normally operated from an external source of power such as from site utilities or a feed from a heavy haul vehicle carrying the HI-TRAC. Occasional interruption in power supply is possible.

11.1.7.2 Detection of SCS Power Failure

The HI-STORM 100 System is designed to withstand a power failure without affecting its ability to meet safety requirements. Consequently SCS monitoring and failure detection is not required.

11.1.7.3 Analysis of Effects and Consequences of SCS Power Failure

The SCS System is required to be equipped with a backup power supply (See SCS specifications in Chapter 2, Appendix 2.C). This ensures uninterrupted operation of the SCS following a power failure. Consequently, a power failure does not effect SCS operation.

Structural

There is no effect on the structural integrity.

Thermal

There is no effect on thermal performance.

Shielding

There is no effect on the shielding performance.

Criticality

There is no effect on the criticality control.

Confinement

There is no effect on the confinement function.

Radiation Protection

As there is no effect on the shielding or confinement functions, there is no effect on occupational or public exposures.

Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.

11.1.7.4 Corrective Action for SCS Power Failure

The HI-STORM 100 System is designed to withstand a power failure without an adverse effect on its normal operation. Consequently no corrective action is required.

11.1.7.5 Radiological Impact of SCS Power Failure

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM 100 System or events postulated because their consequences may affect the public health and safety. Subsection 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM 100 System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. The following sections present the evaluation of the design basis postulated accident conditions and natural phenomena which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for postulated accident conditions are defined in Table 2.2.14. The load combinations include normal loads with the accident loads. The accident load combination evaluations are provided in Section 3.4.

11.2.1 HI-TRAC Transfer Cask Handling Accident

11.2.1.1 Cause of HI-TRAC Transfer Cask Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-TRAC transfer cask is transported to the ISFSI in a vertical position. Unless the lifting device is designed in accordance with the criteria specified in Subsection 2.3.3, the height of the loaded overpack above the ground shall be limited to below the handling height limit determined in Chapter 3 to limit the inertia loading on the cask in a horizontal drop to less than 45g's. Although a handling accident is remote, a cask drop from the horizontal handling height limit is credible only if the lifting device is not designed in accordance with the criteria specified in Subsection 2.3.3. A vertical drop of the loaded HI-TRAC transfer cask is not a credible accident as the loaded HI-TRAC shall be transported and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 and a horizontal drop is precluded as HI-TRAC is transported in the vertical orientation. Nevertheless, for defense-in-depth a drop from a horizontal orientation is postulated and structural consequences evaluated..

11.2.1.2 HI-TRAC Transfer Cask Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded HI-TRAC in the horizontal position. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the HI-TRAC meets all structural requirements and there is no adverse effect on the confinement, thermal or subcriticality performance of the contained MPC. Limited localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket may occur as a result of the handling accident. The HI-TRAC top lid and transfer lid housing (pool lid for the HI-TRAC 125D) are demonstrated to remain attached by withstanding the maximum deceleration. The transfer lid doors (not applicable to HI-TRAC 125D) are also shown to remain closed during the drop. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1]. Therefore, demonstrating that the 45g limit for the HI-TRAC transfer cask is met ensures that the fuel cladding remains intact.

Structural

The structural evaluation of the MPC for 45g's is provided in Section 3.4. As discussed in Section 3.4, the MPC stresses as a result of the HI-TRAC side drop, 45g loading, are all within allowable values.

As discussed above, the water jacket enclosure shell could be punctured which results in a loss of the water within the water jacket. Additionally, the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) are shown to remain in position under the 45g loading. Analysis of the lead in the HI-TRAC is performed in Appendix 3.F and it is shown that there is no appreciable change in the lead shielding.

Thermal

The loss of the water in the water jacket causes the temperatures to increase due to a reduction in the thermal conductivity through the HI-TRAC water jacket. An analysis of the MPC in the HI-TRAC transfer cask temperatures with no water in the water jacket is presented in Subsection 4.6.2. The analysis results are below the short-term allowable fuel cladding and material temperatures limits for accident conditions.

Shielding

The assumed loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded. As the structural analysis demonstrates that the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) remain in place, there is no change in the dose rates at the top and bottom of the HI-TRAC.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

There is no degradation in the confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. The dose rates are provided in Chapter 5. Immediately after the drop accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit exposure . Based on a minimum distance to the controlled area boundary of 100 meters, the 10CFR72.106 dose rate requirements at the controlled area boundary (5 Rem limit) are not exceeded.

11.2.1.3 HI-TRAC Transfer Cask Handling Accident Dose Calculations

The handling accident could cause localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket as the neutron shield impacts the ground.

When the water jacket is impacted, the HI-TRAC transfer cask surface dose rate could increase. The HI-TRAC's post-accident shielding analysis presented in Chapter 5 assumes complete loss of the water in the water jacket and bounds the dose rates anticipated for the handling accident.

If the water jacket of the loaded HI-TRAC is damaged beyond immediate repair and the MPC is not damaged, the loaded HI-TRAC may be unloaded into a HI-STORM overpack, a HI-STAR overpack, or unloaded in the fuel pool. If the MPC is damaged, the loaded HI-TRAC must be returned to the fuel pool for unloading. Depending on the damage to the HI-TRAC and the current location in the loading or unloading sequence, personnel exposure is minimized by continuing to load the MPC into a HI-STORM or HI-STAR overpack. Once the MPC is placed in the HI-STORM or HI-STAR overpack, the dose rates are greatly reduced. The highest personnel exposure will result from returning the loaded HI-TRAC to the fuel pool to unload the MPC.

The analysis of the handling accident presented in Section 3.4 shows that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactive material from the confinement vessel. Any possible rupture of the fuel cladding will have no effect on the site boundary dose rates because the magnitude of the radiation source has not changed.

11.2.1.4 HI-TRAC Transfer Cask Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the HI-TRAC transfer cask and MPC to the maximum practical extent. As appropriate, place temporary shielding around the HI-TRAC to reduce radiation dose rates. Special handling procedures will be developed and approved by the ISFSI operator to lift and upright the HI-TRAC. Upon uprighting, the portion of the overpack not previously accessible shall be radiologically and visually inspected. If damage to the water jacket is limited to a local penetration or crushing, local repairs can be performed to the shell and the water replaced. If damage to the water jacket is extensive, the damage shall be repaired and re-tested in accordance with Chapter 9, following removal of the MPC.

If upon inspection of the damaged HI-TRAC transfer cask and MPC, damage of the MPC is observed, the loaded HI-TRAC transfer cask will be returned to the facility for fuel unloading in accordance with Chapter 8. The handling accident will not affect the ability to unload the MPC using normal means as the structural analysis of the 60g loading (HI-STAR Docket Numbers 71-9261 and 72-1008) shows that there will be no gross deformation of the MPC basket. After unloading, the structural damage of the HI-TRAC and MPC shall be assessed and a determination shall be made if repairs will enable the equipment to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the equipment for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.2 HI-STORM Overpack Handling Accident

11.2.2.1 Cause of HI-STORM Overpack Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-STORM overpack is lifted in the vertical orientation. The height of the loaded overpack above the ground shall be limited to below the vertical handling height limit determined in Chapter 3, unless the lifting device is designed in accordance with the criteria specified in Subsection 2.3.3. This vertical handling height limit will maintain the inertial loading on the cask in a vertical drop to 45g's or less. Although a handling accident is remote, a drop from the vertical handling height limit is a credible accident if the lifting device is not designed in accordance with the criteria specified in Subsection 2.3.3.

11.2.2.2 HI-STORM Overpack Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded overpack in the vertical orientation. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the HI-STORM 100 System meets all structural requirements and there are no adverse effects on the structural, confinement, thermal or subcriticality performance of the HI-STORM 100 System. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1].

Structural

The structural evaluation of the MPC under a 60g vertical load is presented in the HI-STAR FSAR and SAR [11.2.3 and 11.2.4] and it is demonstrated therein that the stresses are within allowable limits. The structural analysis of the HI-STORM overpack is presented in Section 3.4. The structural analysis of the overpack shows that the concrete shield attached to the underside of the overpack lid remains attached and air inlet ducts do not collapse.

Thermal

As the structural analysis demonstrates that there is no adverse effect to the MPC or overpack, there is no effect on the thermal performance of the system as a result of this event.

Shielding

As the structural analysis demonstrates that there is no adverse effect to the MPC or overpack, there is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no adverse effect on the criticality control features of the system as a result of this event.

Confinement

There is no adverse effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the vertical drop of the HI-STORM Overpack with the MPC inside does not affect the safe operation of the HI-STORM 100 System.

11.2.2.3 HI-STORM Overpack Handling Accident Dose Calculations

The vertical drop handling accident of the loaded HI-STORM overpack will not cause any change of the shielding or breach of the MPC confinement boundary. Any possible rupture of the fuel cladding will have no affect on the site boundary dose rates because the magnitude of the radiation source has not changed. Therefore, the dose calculations are equivalent to the normal condition dose rates.

11.2.2.4 HI-STORM Overpack Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, as required, will be developed and approved by the ISFSI operator.

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

11.2.3 Tip-Over

11.2.3.1 Cause of Tip-Over

The analysis of the HI-STORM 100 System has shown that the overpack does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. It is highly unlikely that the overpack will tip-over during on-site movement because of the required low handling height or the use of a lifting device designed in accordance with the criteria specified in Subsection 2.3.3. The tip-over accident is stipulated as a non-mechanistic accident.

For the anchored HI-STORM designs (HI-STORM 100A and 100SA), a tip-over accident is not possible. As described in Chapter 2 of this FSAR, these system designs are not evaluated for the hypothetical tip-over. As such, the remainder of this accident discussion applies only to the non-anchored designs (i.e., the 100 and 100S designs only).

11.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Section 3.4. The structural analysis provided in Appendix 3.A demonstrates that the resultant deceleration loading on the MPC as a result of the tip-over accident is less than the design basis 45g's. The analysis shows that the HI-STORM 100 System meets all structural requirements and there is no adverse effect on the structural, confinement, thermal, or subcriticality performance of the MPC. However, the side impact will cause some localized damage to the concrete and outer shell of the overpack in the radial area of impact.

Structural

The structural evaluation of the MPC presented in Section 3.4 demonstrates that under a 45g loading the stresses are well within the allowable values. Analysis presented in Chapter 3 shows that the concrete shields attached to the underside and top of the overpack lid remains attached. As a result of the tip-over accident there will be localized crushing of the concrete in the area of impact.

Thermal

The thermal analysis of the overpack and MPC is based on vertical storage. The thermal consequences of this accident while the overpack is in the horizontal orientation are bounded by the burial under debris accident evaluated in Subsection 4.6.2. Damage to the overpack will be limited as discussed above. As the structural analysis demonstrates that there is no significant change in the MPC or overpack, once the overpack and MPC are returned to their vertical orientation there is no effect on the thermal performance of the system.

Shielding

The effect on the shielding performance of the system as a result of this event is limited to a localized decrease in the shielding thickness of the concrete.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the accident does not affect the safe operation of the HI-STORM 100 System.

11.2.3.3 Tip-Over Dose Calculations

The tip-over accident could cause localized damage to the radial concrete shield and outer steel shell where the overpack impacts the surface. The overpack surface dose rate in the affected area could increase due to the damage. However, there should be no noticeable increase in the ISFSI site or boundary dose rate, because the affected areas will be small and localized. The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates.

11.2.3.4 Tip-Over Accident Corrective Action

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

If upon inspection of the MPC, structural damage of the MPC is observed, the 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.

11.2.4.2 Fire Analysis

11.2.4.2.1 Fire Analysis for HI-STORM Overpack

The possibility of a fire accident near an ISFSI is considered to be extremely remote due to an absence of combustible materials within the ISFSI and adjacent to the overpacks. The only credible concern is related to a transport vehicle fuel tank fire, causing the outer layers of the storage overpack to be heated by the incident thermal radiation and forced convection heat fluxes. The amount of combustible fuel in the on-site transporter is limited to a volume of 50 gallons.

Structural

As discussed in Section 3.4, there are no structural consequences as a result of the fire accident condition.

Thermal

Based on a conservative analysis discussed in Subsection 4.6.2, of the HI-STORM 100 System response to a hypothetical fire event, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM 100 System to cool the spent nuclear fuel within temperature limits (Table 2.2.3) during and after fire is not compromised.

Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR." Less than one-inch of the concrete (~4% of the overpack radial concrete thickness) exceeds the short-term temperature limit.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100 System.

11.2.4.2.2 Fire Analysis for HI-TRAC Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 100-ton HI-TRAC is performed. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC.

Structural

As discussed in Section 3.4, there are no adverse structural consequences as a result of the fire accident condition.

Thermal

As discussed in Subsection 4.6.2, the MPC internal pressure and fuel temperature increases as a result of the fire accident. The fire accident MPC internal pressure and peak fuel cladding temperature are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).

The thermal analysis of the MPC in the HI-TRAC transfer cask under a fire accident is performed in Subsection 4.6.2. As can be concluded from the analysis, the temperatures for fuel cladding and components are below the accident temperature limits.

Shielding

The assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The assumed loss of all the Holtite in the 125-ton HI-TRAC lids results in an increase in the radiation dose rates at locations adjacent to the lids. The shielding 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, since the internal pressure does not exceed the accident condition design pressure and the MPC confinement boundary temperatures do not exceed the short-term allowable temperature limits.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. Increases in the local dose rates adjacent to the water jacket are evaluated in Chapter 5.. Immediately after the fire accident a radiological inspection of the HI-TRAC shall be performed and temporary shielding shall be installed to limit exposure..

11.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The elevated temperatures experienced by the HI-STORM overpack concrete shield is limited to the outermost layer. Therefore, overall reduction in neutron shielding capabilities is quite small.. The slight increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate, due to the limited amount of the concrete shielding with reduced effectiveness and low site boundary dose rates. The loaded HI-STORM overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC confinement boundary is not compromised and therefore, there is no release of airborne radioactive materials.

11.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC or HI-STORM overpack, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC as the water jacket pressure relief valves may open with resulting water loss and increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, install temporary shielding around the HI-TRAC. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC. If damage to the HI-TRAC is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced . If damage to the HI-TRAC water jacket or HI-TRAC body is widespread and/or radiological conditions require, the HI-TRAC shall be unloaded in accordance with Chapter 8, prior to repair.

If damage to the HI-STORM storage overpack as the result of a fire event is widespread and/or as radiological conditions require, the MPC shall be removed from the HI-STORM overpack in accordance with Chapter 8. However, the thermal analysis described herein demonstrates that only the outermost layer of the radial concrete exceeds its design temperature. The HI-STORM overpack may be returned to service if there is no significant increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the visual inspection is satisfactory.

11.2.5 Partial Blockage of MPC Basket Vent Holes

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. The partial blockage of the MPC basket vent holes analyzes the effects on the HI-STORM 100 System due to the restriction of the vent openings.

11.2.5.1 Cause of Partial Blockage of MPC Basket Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, dried, and backfilled with helium. There are only two possible sources of material that could block the MPC basket vent holes. These are the fuel cladding/fuel pellets and crud. Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture, and that fuel cladding and fuel pellets would fall to block the basket vent holes. It is conceivable that a percentage of the crud deposited on the fuel rods may fall off of the fuel assembly and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. The HI-STORM 100 System maintains the peak fuel cladding temperature below the required long-term storage limits. All credible accidents do not cause the fuel assembly to experience an inertia loading greater than 60g's. Therefore, there is no mechanism for the extensive rupture of spent fuel rod cladding.

Crud can be made up of two types of layers, loosely adherent and tightly adherent. The SNF assembly movement from the fuel racks to the MPC may cause a portion of the loosely adherent crud to fall away. The tightly adherent crud is not removed during ordinary fuel handling operations. The MPC basket vent holes that act as the bottom plenum for the MPC internal thermosiphon are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (i.e., the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. Typically, BWR plants have more crud than PWR plants. Based on the maximum expected crud volume per fuel assembly provided in reference [11.2.2], and the area at the base of the MPC basket fuel storage cell, the maximum depth of crud at the bottom of the MPC-68 was determined. For the PWR-style MPC designs , 90% of the maximum crud volume was used to determine the crud depth. The maximum crud depths for each of the MPCs is listed in Table 2.2.8. The maximum amount of crud was assumed to be present on all fuel assemblies within the MPC. Both the tightly and loosely adherent crud was conservatively assumed to fall off of the fuel assembly. As can be seen by the values listed in the table, the maximum amount of crud depth does not block the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the available vent holes area is greater than that used in the thermal models.

11.2.5.2 Partial Blockage of MPC Basket Vent Hole Analysis

The partial blockage of the MPC basket vent holes has no affect on the structural, confinement and thermal analysis of the MPC. There is no affect on the shielding analysis other than a slight increase of the gamma radiation dose rate at the base of the MPC due to the accumulation of crud. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible, and, therefore, the criticality analyses are not affected.

Structural

There are no structural consequences as a result of this event.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

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

Based on this evaluation, it is concluded that the partial blockage of MPC vent holes does not affect the safe operation of the HI-STORM 100 System.

11.2.5.3 Partial Blockage of MPC Basket Vent Holes Dose Calculations

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary. Therefore, there will be no effect on the site boundary dose rates because the magnitude of the radiation source has not changed. There will be no radioactive material release.

11.2.5.4 Partial Blockage of MPC Basket Vent Holes Corrective Action

There are no consequences that exceed normal storage conditions. No corrective action is required for the partial blockage of the MPC basket vent holes.

11.2.6 Tornado

11.2.6.1 Cause of Tornado

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. Additionally, the transfer of the MPC from the HI-TRAC transfer cask to the overpack may be performed at the unsheltered ISFSI concrete pad. It is possible that the HI-STORM System (storage overpack and HI-TRAC transfer cask) may experience the extreme environmental conditions of a tornado.

11.2.6.2 Tornado Analysis

The tornado accident has two effects on the HI-STORM 100 System. The tornado winds and/or tornado missile attempt to tip-over the loaded overpack or HI-TRAC transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds, which attempt to penetrate the storage overpack or HI-TRAC transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC while being handled in the vertical orientation. During handling of the loaded HI-TRAC in the horizontal orientation, it is possible that the tornado missile and/or wind may cause the rollover of the loaded HI-TRAC on the transport vehicle unless the lifting device is designed in accordance with the requirements specified in Subsection 2.3.3. The horizontal drop handling accident for the loaded HI-TRAC, Subsection 11.2.1, evaluates the consequences of the loaded HI-TRAC falling from the horizontal handling height limit and consequently this bounds the effect of the roll-over of the loaded HI-TRAC on the transport vehicle.

Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the storage overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or tornado missiles.

Analyses provided in Section 3.4 also shows that the tornado missiles do not penetrate the storage overpack or HI-TRAC transfer cask to impact the MPC. The result of the tornado missile impact on the storage overpack or HI-TRAC transfer cask is limited to damage of the shielding.

Thermal

The thermal consequences are evaluated in Subsection 11.2.1.

Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. Increases in the local dose rates as a result of the loss of water in the HI-TRAC water jacket are evaluated in Chapter 5. Immediately after the tornado accident a radiological inspection of the HI-TRAC shall be performed and temporary shielding shall be installed to limit the exposure to the public.

11.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded storage overpack; damage the shielding materials of the overpack or HI-TRAC; or damage the MPC confinement boundary. There is no affect on the radiation dose as a result of the tornado winds. A tornado missile may cause localized damage in the concrete radial shielding of the storage overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

11.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM 100 System to a tornado, the ISFSI operator shall perform a visual and radiological inspection of the overpack and/or HI-TRAC transfer cask. Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and repaired. Damage sustained by the HI-TRAC shall be inspected and repaired.

11.2.7 Flood

11.2.7.1 Cause of Flood

The HI-STORM 100 System will be located on an unsheltered ISFSI concrete pad. Therefore, it is possible for the storage area to be flooded. The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

11.2.7.2 Flood Analysis

Structural

The flood accident affects the HI-STORM 100 overpack structural analysis in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood affects the MPC by applying an external pressure.

Section 3.4 provides the analysis of the flood water applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

Thermal

For a flood of sufficient magnitude to allow the water to come into contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a large flood will enter the annulus between the MPC and the overpack. The water would provide cooling that exceeds that available in the air filled annulus, due to water's higher thermal conductivity.

A smart flood condition that blocks the air flow but is not sufficient to allow water to come into contact with the MPC is bounded by the 100% inlet ducts blocked condition evaluated in Subsection 11.2.13.

Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water provides additional shielding that reduces radiation doses.

Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the fuel pool, which is presented in Section 6.1.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100 System.

11.2.7.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

11.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM 100 System sustains no damage as a result of the flood. At the completion of the flood, exposed surfaces may need debris and adherent foreign matter removal.

11.2.8 Earthquake

11.2.8.1 Cause of Earthquake

The HI-STORM 100 System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM 100 System, the ISFSI may experience an earthquake.

11.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM 100 System. The objective is to determine the stability limits of the HI-STORM 100 System. Based on a static stability criteria, it is shown in Chapter 3 that the HI-STORM 100 System is qualified to seismic activity less than or equal to the values specified in Table 2.2.8. The analyses in Chapter 3 show that the HI-STORM 100 System will not tip over under the conditions evaluated. The seismic activity has no adverse thermal, criticality, confinement, or shielding consequences.

Some ISFSI sites will have earthquakes that exceed the seismic activity specified in Table 2.2.8. For these high-seismic sites, anchored HI-STORM designs (the HI-STORM 100A and 100SA) have been developed. The design of these anchored systems is such that seismic loads cannot result in tip-over or lateral displacement. Chapter 3 provides a detailed discussion of the anchored systems design.

Structural

The sole structural effect of the earthquake is an inertial loading of less than 1g. This loading is bounded by the tip-over analysis presented in Subsection 11.2.3, which analyzes a deceleration of 45g's and demonstrates that the MPC allowable stress criteria are met.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

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

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100 System.

11.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. If the overpack were to tip over, the resultant damage would be equal to that experienced by the tip-over accident analyzed in Subsection 11.2.3. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates or release of radioactivity.

11.2.8.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have tipped-over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 11.2.3.4.

11.2.9 100% Fuel Rod Rupture

This accident event postulates that all the fuel rods rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity.

11.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. There is no credible cause for 100% fuel rod rupture. This accident is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

11.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no thermal, structural, criticality or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, which is being shielded, the shielding capability, or the criticality control features of the HI-STORM 100 System. The determination of the maximum accident pressure is provided in Chapter 4. The MPC design basis internal pressure bounds the pressure developed assuming 100% fuel rod rupture. The structural analysis provided in Chapter 3 evaluates the MPC confinement boundary under the accident condition design internal pressure.

Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

Thermal

The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.9. As can be seen from the values, the design basis accident condition MPC internal pressure (Table 2.2.1) used in the structural evaluation bounds the calculated value.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100 System.

11.2.9.3 100% Fuel Rod Rupture Dose Calculations

The MPC confinement boundary maintains its integrity. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. Therefore, there is no release of radioactive material or significant increase in radiation dose rates.

11.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC confinement boundary is not damaged. The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

11.2.10 Confinement Boundary Leakage

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials for postulated storage accident conditions. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The discussion contained in Chapter 7 demonstrates that MPC is designed, welded, tested and inspected to meet the guidance of ISG-18 such that leakage from the confinement boundary is considered non-credible.

11.2.10.1 Cause of Confinement Boundary Leakage

There is no credible cause for confinement boundary leakage. The accidents analyzed in this chapter show that the MPC confinement boundary withstands all credible accidents. There are no man-made or natural phenomena that could cause failure of the confinement boundary restricting radioactive material release. Additionally, because the MPC satisfies the criteria specified in Interim Staff Guidance (ISG) 18, there is no credible leakage that would occur from the confinement boundary.

11.2.10.3 Confinement Boundary Leakage Accident Corrective Action

The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel. No corrective actions are required.

11.2.11 Explosion

11.2.11.1 Cause of Explosion

An explosion within the bounds of an ISFSI is improbable since there are no explosive materials within the site boundary. An explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity the effects of an explosion on a reinforced structure are minimal. Explosions postulated to occur within or beyond the ISFSI boundary would require a site hazards evaluation under the provisions of 72.212 regulations by individual cask users.

11.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external design pressure (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 11.2.7 and the tornado missile accident of Subsection 11.2.6, because explosive materials are not stored within close proximity to the casks. The bounding analysis shows that the MPC and the Overpack can withstand the effects of substantial accident external pressures without collapse or rupture.

Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100 System.

11.2.11.3 Explosion Dose Calculations

The bounding external pressure load has no effect on the HI-STORM 100 overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM 100 System is experienced as a result of the explosion pressure load. The effects of explosion generated missiles on the HI-STORM 100 System structure is bounded by the analysis of tornado generated missiles.

11.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by the explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM 100 System. Following an explosion, the ISFSI operator shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the concrete material may be replaced and the outer shell repaired.

11.2.12 Lightning

11.2.12.1 Cause of Lightning

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad. There is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

11.2.12.2 Lightning Analysis

The HI-STORM 100 System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM 100 System is hit with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, will provide a direct path to ground.

The MPC provides the confinement boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

Structural

There is no structural consequence as a result of this event.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

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

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STORM 100 System.

11.2.12.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the confinement boundary or shielding materials. Therefore, no further analysis is necessary.

11.2.12.4 Lightning Accident Corrective Action

The HI-STORM 100 System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

11.2.13 100% Blockage of Air Inlets

11.2.13.1 Cause of 100% Blockage of Air Inlets

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

11.2.13.2 100% Blockage of Air Inlets Analysis

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

As a result of the large mass, and correspondingly large thermal capacity of the storage overpack, 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.

Structural

There are no structural consequences as a result of this event.

Thermal

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

Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

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

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

11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets accident, the shielding capabilities of the HI-STORM 100 System are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within 32 hours. 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 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 doses. The accident external pressure encountered during the flood bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STORM System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limit during a burial under debris accident.

Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100 System, if the debris is removed within the specified time. The 24-hour minimum duct inspection interval ensures that a burial under debris condition will be detected long before the allowable burial time is reached.

11.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

11.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures are not exceeded even for an extended duration of burial. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be performed prior to the reinsertion of the MPC. The site's emergency action plan shall include provisions for the performance of this corrective action.

11.2.15 Extreme Environmental Temperature

11.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

11.2.15.2 Extreme Environmental Temperature Analysis

The accident condition considering an extreme environmental temperature (Table 2.2.2) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3.

Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in evaluation performed in Subsection 4.6.2. As concluded from this evaluation, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100 System.

11.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature and the dose calculations are equivalent to the normal condition dose rates.

11.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

11.2.16 Supplemental Cooling System (SCS) Failure

The SCS system is a forced fluid circulation device used to provide supplemental HI-TRAC cooling. For fluid circulation, the SCS system is equipped with active components requiring power for normal operation. Although an SCS System failure is highly unlikely, for defense-in-depth an accident condition that renders it inoperable for an extended duration is postulated herein.

11.2.16.1 Cause of SCS Failure

Possible causes of SCS failure are: (a) Simultaneous loss of external and backup power, or (b) Complete loss of annulus water from an uncontrolled leak or line break.

11.2.16.2 Analysis of Effects and Consequences of SCS Failure

Structural

See discussion under thermal evaluation below.

Thermal

In the event of a SCS failure due to (a), the following sequence of events occur:

i) The annulus water temperature rises to reach it's boiling temperature (~212°F).
ii) A progressive reduction of water level and dryout of the annulus.

In the event of an SCS failure due to (b), a rapid water loss occurs and annulus is replaced with air. For the condition of a vertically oriented HI-TRAC with air in the annulus, the maximum steady-state temperatures are below the accident temperature limits for fuel cladding and components (see Subsection 4.5.4).

Shielding

There is no adverse effect on the shielding effectiveness of the system.

Criticality

There is no adverse effect on the criticality control of the system.

Confinement

There is no adverse effect on the confinement function of the MPC. As discussed in the evaluations above, the structural boundary pressures are within design limits.

Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the SCS failure does not affect the safe operation of the HI-STORM 100 System.

11.2.16.3 SCS Failure Dose Calculations

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.2.16.4 SCS Failure Corrective Action

In the vertical orientation the HI-TRAC is designed to withstand an SCS failure without an adverse effect on its safety functions.

11.3 REFERENCES

[11.2.1] Chun et al., "Dynamic Impact Effects on Spent Fuel Assemblies," Lawrence Livermore National Laboratory, UCID-21246, (October 1987).

[11.2.2] ESEERCO Project EP91-29 and EPRI Project 3100-02, "Debris Collection System for Boiling Water Reactor Consolidation Equipment," B&W Fuel Company, (October 1995).

[11.2.3] Docket Number 72-1008, HI-STAR 100 System FSAR, Holtec Report HI-2012610, Revision 2.

[11.2.4] Docket Number 71-9261, HI-STAR 100 System SAR, Holtec Report HI-951251, Revision 10.

SUPPLEMENT 11.I

ACCIDENT EVALUATION FOR THE HI-STORM 100U SYSTEM

11.I.0 INTRODUCTION

This supplement is focused on the off-normal and accident condition evaluations of the HI-STORM 100U vertical ventilated module (VVM). Only those events that are actually affected by the design of the overpack are discussed in detail herein. The reader is referred to the main body of Chapter 11 for discussions of any off-normal or accident conditions that are not dependent on the design of the storage overpack (i.e., MPC-only or HI-TRAC events).

The evaluations described herein parallel those of the HI-STORM 100 overpack contained in the main body of Chapter 11 of this FSAR. To ensure readability, the sections in this supplement are numbered to be directly analogous to the sections in the main body of the chapter. For example, the fire accident evaluation presented in Supplement Subsection 11.I.2.4 for the HI-STORM 100U is analogous to the evaluation presented in Subsection 11.2.4 of the main body of Chapter 11 for the HI-STORM 100. Tables and figures (if any) in this supplement, however, are labeled sequentially by section. If there is an analogous table or figure in the main body of Chapter 11, an appropriate notation is made in the supplement table or figure.

11.I.1 OFF-NORMAL EVENTS

A general discussion of off-normal events is presented in Section 11.1 of the main body of Chapter 11. The following off-normal events are discussed in this supplement:

Off-Normal Pressure
Off-Normal Environmental Temperature
Leakage of One MPC Seal Weld
Partial Blockage of Air Inlets
Off-Normal Handling of HI-TRAC Transfer Cask
Malfunction of FHD System
SCS Power Failure
Off-Normal Wind

The results of the evaluations presented herein demonstrate that the HI-STORM 100U System can withstand the effects of off-normal events without affecting its ability to perform its intended function, and is in compliance with the applicable acceptance criteria.

11.I.1.1 Off-Normal Pressure

A discussion of this off-normal condition is presented in Subsection 11.1.1 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The applicable pressure boundary stress limits are confirmed to bound the stresses resulting from the off-normal pressure.

Thermal

In 4.6.1 the MPC internal pressure under the conditions of 10% fuel rods ruptured, insolation and a limiting fuel storage configuration in an aboveground overpack is evaluated. This evaluation is bounding as the MPC temperatures in the 100U overpack are bounded by the aboveground overpack.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation mentioned above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100U System.

11.I.1.2 Off-Normal Environmental Temperatures

A discussion of this off-normal condition is presented in Subsection 11.1.2 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. However,as shown previously the resultant pressure is below the off-normal design pressure (Table 2.2.1). The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsections 3.1.2 and 3.I.1.

Thermal

Supplement 4.I calculates bounding temperatures and pressures for the HI-STORM 100U under the elevated temperature condition. The calculated temperatures and pressures are reported in Table 4.I.3 and are below the off-normal limits (Tables 2.2.3, 2.I.7 and 2.2.1).

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM 100U overpack. The HI-STORM 100U overpack is conservatively assumed to reach -40°F throughout the structure. Chapter 3, Subsection 3.1.2 details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100U System.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100U System.

11.I.1.3 Leakage of One MPC Seal Weld

A discussion of this off-normal condition is presented in Subsection 11.1.3 of the main body of Chapter 11. The discussion presented therein is applicable in it's entirety to an MPC in a HI-STORM 100U VVM as well.

11.I.1.4 Partial Blockage of Air Inlets

A discussion of this off-normal condition is presented in Subsection 11.1.4 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

Supplement 4.I calculates bounding temperatures for 50% blockage of the air inlets. The calculated bounding temperatures are reported in Table 4.I.5 and are below the MPC and VVM off-normal design temperatures (Tables 2.2.3 and 2.I.7). Additionally, the increased temperatures generate an elevated MPC internal pressure, also reported in Table 4.I.4, which is less than the off-normal design pressure (Table 2.2.1).

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100U System.

11.I.1.5 Off-Normal Handling of HI-TRAC

A discussion of this off-normal condition is presented in Subsection 11.1.5 of the main body of Chapter 11. The discussion presented therein remains completely applicable, as the design and method of operation of the HI-TRAC is the same as with the HI-STORM 100U.

11.I.1.6 Failure of FHD System

A discussion of this off-normal condition is presented in Subsection 11.1.6 of the main body of Chapter 11. The discussion presented therein remains completely applicable for all MPCs.

11.I.1.7 SCS Power Failure

A discussion of this off-normal condition is presented in Subsection 11.1.7 of the main body of Chapter 11. The discussion presented therein remains completely applicable to all MPCs.

11.I.1.8 Off-Normal Wind

The HI-STORM 100U is designed for use at any site in the United States. Supplement 4.I evaluates the effects of off-normal wind (between 10 and 30 MPH). The off-normal wind is postulated as a constant horizontal wind caused by extreme weather conditions (see Table 2.I.1). To determine the effects of the off-normal wind, it is conservatively assumed that these winds persist for a sufficient duration to allow the HI-STORM 100U System to reach thermal equilibrium. Because of the large mass of the HI-STORM 100U System with its corresponding large thermal inertia and the unlikely condition of a unidirectional wind for a long period of time, this assumption is conservative. The analyses presented in Supplement 4.I shows that the peak fuel cladding and material temperatures remains below the off-normal limits (Tables 2.2.3 and 2.I.7). Because the HI-STORM 100U System is designed to withstand the off-normal wind without any effect on its ability to maintain safe storage conditions, there is no requirement for detection of the off-normal wind.

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

Supplement 4.I calculates peak fuel cladding temperatures for horizontal wind speeds of up to 30 miles per hour. The calculated temperatures (reported in Table 4.I.9) are below the off-normal limits (Table 2.2.3).

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal wind event does not affect the safe operation of the HI-STORM 100U System. The HI-STORM 100U System is designed to withstand the off-normal wind without any effect on its ability to maintain safe storage conditions. There are no corrective actions required for the off-normal wind. The off-normal wind has no radiological impact, and the confinement barrier and shielding integrity are not affected.

11.I.2 ACCIDENT EVENTS

A general discussion of accident events is presented in Section 11.1 of the main body of Chapter 11. The following accident events are discussed in this supplement section:

HI-TRAC Transfer Cask Handling Accident
HI-STORM 100U Overpack Handling Accident
Tip-Over
Fire Accident
Partial Blockage of MPC Basket Vent Holes
Tornado
Flood
Earthquake
100% Fuel Rod Rupture
Confinement Boundary Leakage
Explosion
Lightning
100% Blockage of Air Inlets
Burial Under Debris
Extreme Environmental Temperature
SCS Failure

The results of the evaluations performed herein demonstrate that the HI-STORM 100U System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and is in compliance with the applicable acceptance criteria.

11.I.2.1 HI-TRAC Transfer Cask Handling Accident

A discussion of this accident condition is presented in Subsection 11.2.1 of the main body of Chapter 11. The discussion presented therein is applicable in it's entirety, as the design and method of operation of the HI-TRAC is the same for the HI-STORM 100U.

11.I.2.2 HI-STORM Overpack Handling Accident

This accident event is not applicable to the HI-STORM 100U as this is an underground overpack secured at the base with anchors embedded in a thick concrete pad and surrounded by soil.

11.I.2.3 Tip-Over

This accident event is not applicable to the HI-STORM 100U. Due to the subterranean anchored installation of the VVM with a surrounding subgrade for lateral support, tip-over is precluded.

11.I.2.4 Fire Accident

A discussion of this accident condition is presented in Subsection 11.2.4 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein. In addition, the discussion of the fire analysis for the HI-TRAC transfer cask presented therein remains completely applicable, as the design and method of operation of the HI-TRAC do not need to be changed for use with the HI-STORM 100U.

Structural

There are no structural consequences as a result of the fire accident condition.

Thermal

Supplement 4.I discusses the impact of a fire on the HI-STORM 100U System. As justified therein, the evaluation for the fire effects on an aboveground cask presented in Section 11.2 bound the effects on the HI-STORM 100U System. As described in Section 11.2, the effects of the fire do not cause any system component or the contained fuel to exceed any design limit. As such, the results are bounding for the HI-STORM 100U System.

Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR."

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100U System.

11.I.2.5 Partial Blockage of MPC Basket Vent Holes

A discussion of this accident condition is presented in Subsection 11.2.5 of the main body of Chapter 11. The discussion presented therein is applicable in it's entirety to an MPC in a HI-STORM 100U VVM.

11.I.2.6 Tornado

A discussion of this accident condition is presented in Subsection 11.2.6 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Because of its underground construction, the HI-STORM 100U is not affected by the tornado wind. The effect of tornado missiles propelled by high velocity winds that attempt to penetrate the exposed portions of the HI-STORM 100U must, however, be considered.

Structural

Analyses presented in Supplement 3.I show that the impact of an intermediate tornado missile on the HI-STORM 100U closure lid does not result in the perforation of the lid or result in a structural collapse. The result of the tornado missile impact on the VVM is limited to localized damage of the shielding.

Thermal

There are no thermal consequences as a result of the tornado beyond those discussed for the wind herein.

Shielding

A tornado missile may cause localized damage to the HI-STORM 100U closure lid. As the HI-STORM 100U top is heavily shielded (a thick MPC lid backed up by a steel-concrete-steel top) the overall damage consequences (site boundary dose) are insignificant.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. A tornado missile may cause localized damage in the HI-STORM 100U closure lid. However, the damage will have a negligible effect on the site boundary dose.

Based on this evaluation, it is concluded that the tornado accident does not affect the safe operation of the HI-STORM 100U System.

11.I.2.7 Flood

A discussion of this accident condition is presented in Subsection 11.2.7 of the main body of Chapter 11. A description of the cause of this event is presented therein.

Structural

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

Thermal

The thermal consequences of flood are evaluated in Supplement 4.I.4. The calculated bounding temperatures are reported in Table 4.I.5. The results are below the accident temperature limits (Table 2.2.3 and 2.I.7). The results demonstrate that water intrusion in the VVM cooling passages under all limiting scenarios yields temperatures below the allowable limits. Safe operation of the HI-STORM 100U System is assured.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event. The floodwater provides additional shielding which reduces radiation dose.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the spent fuel pool, which is presented in Section 6.1.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100U System.

Flood Accident Corrective Action

As discussed in Supplement 1.I,the configuration of the VVM makes it uniquely suited to withstand a flooding event .Indeed ,introducing water in the CEC is an effective method to lower the MPC contents' temperature. However, solid debris packed around the Divider shell is an undesirable condition. Thus, while the thermal evaluations discussed in Supplement 4.I demonstrate that the HI-STORM 100U System will safely withstand a flood, corrective actions after such an event may be necessary. Periodic VVM air temperature monitoring, required for the HI-STORM 100U System, will identify any blockage of the cooling passages that results in a non-normal thermal condition, including blockages due to a flood borne debris.

If the measured temperature rise exceeds the allowable value, then corrective actions to alleviate the condition will be required. To restore the system to a normal configuration, all flood water and any debris deposited by the receding water must be removed. The specific methods to be used are appropriately site specific and shall be addressed in the site emergence action plan. Examples of acceptable cleaning approaches include:

1. The MPC is removed from the VVM using the HI-TRAC transfer cask, allowing direct access to the interior of the VVM through both the inlet vents and the top of the module cavity. Water sprays and vacuuming is used to directly clean the VVM passages and surfaces.

2. Appropriate vacuuming equipment is inserted through the inlet ducts and down to the bottom plenum. Water is sprayed in through the outlet vents. Remote cameras are used to inspect the VVM cooling passages to identify debris and remove debris.

The adequacy of the cooling passages clearance operation is verified by visual inspection or, if the optional air temperature monitoring is used, return of the air outlet temperatures to within allowable limits.

11.I.2.8 Earthquake

A discussion of this accident condition is presented in Subsection 11.2.8 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

Because of its underground construction, the HI-STORM 100U VVM is inherently safe under seismic events. Analyses presented in Supplement 3.I show that the VVM will continue to render its intended function under a seismic event whose ZPAs are bounded by the values set forth in Supplement 2.I .

Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

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

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100U System.

11.I.2.9 100% Fuel Rod Rupture

A discussion of this accident condition is presented in Subsection 11.2.9 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

Thermal

A bounding MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.9. The design basis accident condition MPC internal pressure (Table 2.2.1) used in the structural evaluation bounds the calculated value.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100U System.

11.I.2.10 Confinement Boundary Leakage

A discussion of this accident condition is presented in Subsection 11.2.10 of the main body of Chapter 11. The discussion presented therein remains completely applicable to an MPC in a HI-STORM 100U VVM as well.

11.I.2.11 Explosion

A discussion of this accident condition is presented in Subsection 11.2.11 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

Because of its underground construction, the HI-STORM 100U and the MPC contained within are essentially shielded by the surrounding earth. Thus, no evaluation of the VVM or the contained MPC is required. The HI-STORM 100U closure lid is, however, aboveground and exposed to the explosion-induced pressure wave. Supplement 3.I includes an evaluation of the effect of the design-basis 10 psi pressure wave applied as a static pressure on the closure lid. This evaluation shows that the overpressure wave does not result in lid separation, and that all lid stresses are a fraction of the allowable limits.

Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain well within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100U System.

11.I.2.12 Lightning

A discussion of this accident condition is presented in Subsection 11.2.12 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Because of its underground construction, the subterranean portion of the HI-STORM 100U would not be subjected to a direct lightning strike. The HI-STORM 100U closure lid is, however, aboveground and could be subjected to a direct strike. The closure lid is, however, a steel encased concrete structure just like on the aboveground casks. Thus, the discussion presented in Subsection 11.2.12 remains completely applicable to the exposed portions of the HI-STORM 100U System.Therefore, it is concluded that a lightning event will not prevent the VVM from rendering its intended function.

11.I.2.13 100% Blockage of Air Inlets

A discussion of this accident condition is presented in Subsection 11.2.13 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

There are no structural consequences as a result of this accident event.

Thermal

Supplement 4.I calculates bounding temperatures for the 100% blockage of the air inlets. The calculated bounding temperatures after 26 hours of 100% blockage are reported in Table 4.I.6. The results are below the MPC and VVM accident temperature limits (Tables 2.2.3 and 2.I.7).

Additionally, the increased temperatures generate an elevated MPC internal pressure, also reported in Table 4.I.6, which is less than the design basis accident pressure listed in Table 2.2.1.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperatures do not exceed the accident temperature limit.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

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

Based on this evaluation, it is concluded that the 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.I.2.14 Burial Under Debris

A discussion of this accident condition is presented in Subsection 11.2.14 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

Thermal

Supplement 4.I discusses the impact of burial under debris on the HI-STORM 100U System. As explained therein, the evaluation for the effects of such an event on an aboveground cask presented in Section 11.2 bound the HI-STORM 100U.

Shielding

There is no adverse effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100U System, if the debris is removed within the specified time period.

11.I.2.15 Extreme Environmental Temperature

A discussion of this accident condition is presented in Subsection 11.2.15 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by the design-basis internal pressure and are well within the allowable values, as discussed in Section 3.4.

Thermal

Supplement 4.I calculates bounding temperatures for the HI-STORM 100U under the extreme environmental temperature condition. The calculated bounding temperatures and pressures are reported in Table 4.I.7 and are below the MPC and VVM accident temperature and pressure limits (Tables 2.2.3, 2.I.7 and 2.2.1).

Shielding

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

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

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100U System.

11.I.2.16 Supplemental Cooling System (SCS) Failure

A discussion of this off-normal condition is presented in Subsection 11.2.16 of the main body of Chapter 11. The discussion presented therein remains completely applicable, as the design and method of operation of the SCS and the HI-TRAC is unchanged for use with the HI-STORM 100U System.

12.1 PROPOSED OPERATING CONTROLS AND LIMITS

12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

12.1.1.1 This portion of the FSAR establishes the commitments regarding the HI-STORM 100 System and its use. Other 10CFR72 [12.1.2] and 10CFR20 [12.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [12.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM 100 System. The general license conditions governed by 10CFR72 [12.1.2] are not repeated with these Technical Specifications. Licensees are required to comply with all commitments and requirements.

12.1.1.2 The Technical Specifications provided in Appendix A to CoC 72-1014 and the authorized contents and design features provided in Appendix B to CoC 72-1014 are primarily established to maintain subcriticality, confinement boundary and intact fuel cladding integrity, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 12.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 12.1.2 provides the list of Technical Specifications for the HI-STORM 100 System.

Table 12.1.1

HI-STORM 100 SYSTEM CONTROLS

Condition to be Controlled	**Applicable Technical Specifications†**
Criticality Control	3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) 3.1.4 Supplemental Cooling System
Shielding and Radiological Protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down 3.2.2 TRANSFER CASK Surface Contamination *5.4 Radioactive Effluent Control Program* 5.7 Radiation Protection Program
Heat Removal Capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System 3.1.4 Supplemental Cooling System
Structural Integrity	~~3.5 Cask Transfer Facility (CTF)~~ 5.5 Cask Transport Evaluation Program

† Technical Specifications are located in Appendix A to CoC 72-1014. Authorized contents are specified in FSAR Section 2.1.9

Table 12.1.2

HI-STORM 100 SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION 1.1 Definitions 1.2 Logical Connectors 1.3 Completion Times 1.4 Frequency
2.0	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.1.4	Supplemental Cooling System
3.2.1	Deleted
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	Deleted
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINSTRATIVE CONTROLS ~~AND PROGRAMS~~
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	Deleted
5.7	Radiation Protection Program
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements

12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR.

12.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM 100 Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM 100 System Design (overview);

2. ISFSI Facility Design (overview);

3. Systems, Structures, and Components Important to Safety (overview);

4. HI-STORM 100 System Final Safety Analysis Report (overview);

5. NRC Safety Evaluation Report (overview);

6. Certificate of Compliance conditions;

7. HI-STORM 100 Technical Specifications, Approved Contents, Design Features and other Conditions for Use;

8. HI-STORM 100 Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);

9. Required instrumentation and use;

10. Operating Experience Reviews

11. HI-STORM 100 System and ISFSI Procedures, including

    - Procedural overview
    - Fuel qualification and loading
    - MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
    - MPC welding operations
    - HI-TRAC/overpack closure
    - Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, supplemental cooling, and cooldown)

- MPC/HI-TRAC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

12.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM 100 System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The dry run shall include, but is not limited to the following:

1. Receipt inspection of HI-STORM 100 System components.

2. Moving the HI-STORM 100 MPC/HI-TRAC into the spent fuel pool.

3. Preparation of the HI-STORM 100 System for fuel loading.

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

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

6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.

7. Replacing the HI-TRAC pool lid with the transfer lid (HI-TRAC 100 and 125 only).

8. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup may be used).

9. HI-TRAC upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.

10. Placement of the HI-STORM 100 System at the ISFSI.

11. HI-STORM 100 System unloading, including cooling fuel assemblies, flooding the MPC cavity, and removing MPC welds (for which a mock-up may be used).

12. Installation and operation of the Supplemental Cooling System.

12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STORM 100 System is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system to verify operability of the overpack heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

12.2.4 Limiting Conditions for Operation

Limiting Conditions for Operation specify the minimum capability or level of performance that is required to assure that the HI-STORM 100 System can fulfill its safety functions.

12.2.5 Equipment

The HI-STORM 100 System and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown in this FSAR that no credible condition or event prevents the HI-STORM 100 System from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

12.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM 100 System fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Section 2.1.9 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

12.2.7 Design Features

This section describes HI-STORM 100 System design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in this FSAR and in Appendix B to CoC 72-1014, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM 100 System is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 9.

12.2.8 MPC

a. Basket material composition, properties, dimensions, and tolerances for criticality control.

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

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

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

12.2.9 HI-STORM Overpack

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

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

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

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

The~~is~~ example*s* *below* executes the methodology and equations described in Section 2.1.9.1 for determining allowable decay heat, burnup, and cooling time for the approved cask contents.~~In this example a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern.~~

~~Step 1: Determine the maximum allowable assembly decay heat load for each region.~~

~~$q_{Region 1}$ = 1.131 kW~~
~~$q_{Region 2}$ = 0.600 kW~~

~~Step 2: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 using Equation 2.1.9.3.~~

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

~~Fuel Assembly Number 1 is not acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.1 and 12.1.2. The solution is to develop~~

~~another table using an enrichment of 3.0 wt.% ^{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 4.6 years is calculated by linear interpolation between the values in Table 12.2.1 for 4 years and 5 years of cooling. Linear interpolation yields a maximum burnup of 39,843 MWD/MTU (rounded down from 39,843.4), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.~~

~~Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment 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 1

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

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

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

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

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

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

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

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† *Results are arbitrarily rounded to four decimal places.*

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.

<u>Example 2</u>

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern. Assuming the same value 'X', the same maximum fuel storage location decay heats are calculated. 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).

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

Table 12.2.1

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-32, Array/Class 16x16A, X = *2.8*, and Enrichment = 3.1 wt.% ^{235}U)
($q_{Region\ 1}$ = *1.5871*~~1.131~~ kW, $q_{Region\ 2}$ = 0.*5668*~~600~~ kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	~~24432~~*32791*	~~12303~~*10896*
≥4	~~35110~~*45145*	~~19318~~*17370*
≥5	~~42999~~*53769*	~~24991~~*22697*
≥6	~~48530~~*59699*	~~29209~~*26615*
≥7	~~52394~~*63971*	~~32135~~*29386*
≥8	~~55322~~*67343*	~~34318~~*31437*
≥9	~~57636~~*68200*	~~36005~~*33000*
≥10	~~59584~~*68200*	~~37395~~*34271*
≥11	~~61262~~*68200*	~~38552~~*35384*
≥12	~~62786~~*68200*	~~39584~~*36322*
≥13	~~64206~~*68200*	~~40507~~*37189*
≥14	~~65551~~*68200*	~~41368~~*37980*
≥15	~~66881~~*68200*	~~42200~~*38773*
≥16	~~68184~~*68200*	~~42998~~*39512*
≥17	68200	~~43769~~*40234*
≥18	68200	~~44538~~*40908*
≥19	68200	~~45292~~*41620*
≥20	68200	~~46055~~*42324*

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*~~131~~ kW, $q_{Region 2}$ = 0.*5668*~~600~~ kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	~~25811~~*34797*	~~12639~~*11101*
≥4	~~36903~~*47590*	~~19962~~*17870*
≥5	~~44965~~*56438*	~~25702~~*23272*
≥6	~~50602~~*62533*	~~29910~~*27157*
≥7	~~54568~~*66963*	~~32830~~*29907*
≥8	~~57592~~*68200*	~~35020~~*31935*
≥9	~~59984~~*68200*	~~36710~~*33510*
≥10	~~62016~~*68200*	~~38132~~*34785*
≥11	~~63766~~*68200*	~~39321~~*35927*
≥12	~~65351~~*68200*	~~40372~~*36894*
≥13	~~66822~~*68200*	~~41330~~*37790*
≥14	68200	~~42224~~*38593*
≥15	68200	~~43086~~*39419*
≥16	68200	~~43913~~*40191*
≥17	68200	~~44698~~*40937*
≥18	68200	~~45497~~*41643*
≥19	68200	~~46279~~*42363*
≥20	68200	~~47067~~*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	~~0.7~~*1.01*	BPRA	0.5~~3~~
2	3.2	~~38812~~*35250*	~~4.6~~*3.3*	~~0.9~~*1.45*	NA	NA
3	4.3	~~41976~~*41276*	18.2	0.4	BPRA	0.1

HI-STORM 100 SYSTEM FSAR

APPENDIX 12.A

TECHNICAL SPECIFICATION BASES

FOR THE HOLTEC HI-STORM 100 SPENT FUEL STORAGE CASK SYSTEM

~~**(46 PAGES, INCLUDING THIS PAGE)**~~

BASES TABLE OF CONTENTS

B 3.1 SFSC Integrity

B 3.1.1 Multi-Purpose Canister (MPC)

BASES

BACKGROUND

A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain port cover plates and closure ring are installed and welded. Inspections are performed on the welds.

MPC cavity moisture removal using vacuum drying or forced helium dehydration is performed to remove residual moisture from the MPC cavity space after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel and by the heat added to the MPC from the optional warming pad, if used.

If forced helium dehydration is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

(continued)

BASES

BACKGROUND (continued)

After the completion of drying, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Backfilling the MPC with helium in the required quantity eliminates air inleakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage.

~~In-leakage of air could be harmful to the fuel. Prior to moving the SFSC to the storage pad, the MPC helium leak rate is determined to ensure that the fuel is confined.~~

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC overpressurization evaluation.

(continued)

BASES (continued)

LCO	A dry, helium filled and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the multiple confinement boundaries. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.
APPLICABILITY	The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure both the confinement barriers and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored fuel.
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A.1 If the cavity vacuum drying pressure or demoisturizer exit gas temperature limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS (continued)

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the helium backfill quantity limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS (continued)

B.2

Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

C.1

If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid (if required), perform fuel cooldown operations (if required), re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 and SR 3.1.1.2

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. For moderate burnup fuel cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the gas temperature or dew point at the specified location reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. For high burnup fuel and high decay heat load MPCs, the forced helium dehydration method of moisture removal must be used to provide necessary cooling of the fuel during drying operations. Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations.

Table 3-1 of Appendix A to the CoC provides the appropriate requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high) and the applicable short-term temperature limit. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11.

Having the proper quantity of helium in the MPC ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 and SR 3.1.1.2 (continued) Both of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the cask design.
REFERENCES	1. FSAR Sections 1.2, 4.4, 4.5, 7.2, 7.3 and 8.1 2. Interim Staff Guidance Document 11 3. Interim Staff Guidance Document 18

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES

BACKGROUND	The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system that ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the inlet air ducts ~~at the bottom of the OVERPACK~~. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the outlet air ducts at the top of the OVERPACK.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the inlet and outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs. Analyses have been performed for the complete obstruction of half, and all inlet air ducts. Blockage of half of the inlet air ducts reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The complete blockage of all inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler ~~inner shell of the~~ OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the ~~72-hour~~ duration of the analyzed event.

LCO

The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. *Operability is defined as at least 50% of the inlet air ducts available for air flow (i.e., unblocked).* Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.

The intent of this LCO is to address those occurrences of air duct blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris).

(continued)

BASES

LCO
(continued)

This LCO is not intended to address low frequency, unexpected Design Event III and IV class events such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.109 of Appendix B to the CoC.

APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate dissipation of the decay heat from the fuel assemblies.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

(continued)

BASES

ACTIONS (continued)

B.1

If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, 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.

B.2.1

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

For an aboveground OVERPACK, ~~T~~this Required Action must be complete in 64 hours *if the decay heat load of the MPC is less than or equal to 28.74 kW or within 24 hours if the decay heat load of the MPC is greater than 28.74 kW. For an underground OVERPACK, this required action must be complete in 18 hours. 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 26 hours after event initiation, respectively.*

(continued)

BASES

ACTIONS

B.2.1 (continued)

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

B.2.2

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

An engineering evaluation must be performed to determine if any concrete deterioration has occurred which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

(continued)

BASES

ACTIONS

B.2.2 (continued)

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Times of 64, *24 and 18* hours reflects the Completion Time from Required Action B.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads *and OVERPACK styles.*

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. ~~Complete~~*Greater than 50%* blockage of *the total* ~~any one or more~~ inlet or outlet air ducts *area* renders the heat removal system inoperable and this LCO not met. ~~Partial~~*50% or less* blockage of *the total*~~one or more~~ inlet or outlet air ducts *area* does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limit stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency of 24 hours *for aboveground OVERPACKs and 18 hours for underground OVERPACKs* is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.

REFERENCES

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

B 3.1 SFSC Integrity

B 3.1.4 Supplemental Cooling System

BASES

BACKGROUND	The Supplemental Cooling System (SCS) is an active, water cooling system that provides augmented heat removal from the MPC to ensure fuel cladding temperatures remain below the applicable limit during onsite transport operations in the TRANSFER CASK. The system is required for all MPCs meeting the burnup, heat load, and TRANSFER CASK orientation combinations specified in the Applicability of the LCO.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the MPC inside the TRANSFER CASK take credit for the operation of the SCS under certain conditions to ensure that the spent fuel cladding temperature remains below the applicable limit. FSAR Section 4.5 describes these analyses in more detail. For MPCs containing all moderate burnup fuel ($\leq$ 45,000 MWD/MTU), SCS operation is not required, because the fuel cladding temperature cannot exceed the limit of 1058°F for moderate burnup fuel (Refs. 2 and 3). For high burnup fuel, the fuel cladding temperature limit is 400°C (752°F) during onsite transportation. For MPCs containing one or more high burnup fuel assemblies, the SCS has been credited in the thermal analysis in order to meet the lower fuel cladding temperature limit.

(continued)

BASES

LCO

The Supplemental Cooling System must be operable if the MPC/TRANSFER cask assemblage meets one of the following conditions in the Applicability portion of the LCO in order to preserve the assumptions made in the thermal analysis.

APPLICABILITY

The LCO is applicable within 4 hours after completion of MPC drying operations in accordance with LCO 3.1.1 or within 4 hours of transferring the MPC into the TRANSFER CASK if the MPC is to be unloaded, and the following conditions are met:

MPCs having one or more fuel assemblies with an average burnup greater than 45,000 MWD/MTU.

MPCs having a decay heat load > 25 kW.

ACTIONS

A.1

The SCS would be declared inoperable if it cannot be operated due to a failure of a system component or a TRANSFER CASK component, or the inaccessibility of a portion of the transfer cask (i.e., the bottom lid fluid connection). If the SCS has been determined to be inoperable, the thermal analysis shows that the fuel cladding temperature would not exceed the short term temperature limit applicable to an off-normal condition, even with no water in the TRANSFER CASK-to-MPC annulus. Actions should be taken to restore the SCS to operable status in a timely manner. Because the thermal analysis is a steady-state analysis, there is an indefinite period of time available to make repairs to the SCS. However, it is prudent to require the actions to be completed in a reasonably short period of time. A Completion Time of 7 days is considered appropriate and a reasonable amount of time to plan the work, obtain needed parts, and execute the work in a controlled manner.

(continued)

BASES

ACTIONS (continued)

B.1

If, after 7 days, the SCS cannot be restored to operable status, actions should be taken to remove the fuel assemblies from the MPC and place them back into the spent fuel pool storage racks. Thirty days is considered a reasonable time frame given that the MPC will be adequately cooled while this action is being planned and implemented, and certain equipment for this infrequent evolution (e.g., weld cutting machine) may take some time to acquire.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment, including during short-term evolutions such as on-site transportation in the TRANSFER CASK. The SCS is required to ensure adequate fuel cooling in certain cases. The SCS should be verified to be operable every two hours. This would involve verification that the water flow rate and temperatures are within expected ranges and the pump and air cooler are operating as expected. This is a reasonable Frequency given the typical oversight occurring during the on-site transportation evolution, the duration of the evolution, and the simple equipment involved.

REFERENCES

1. FSAR Section 4.5
2. NRC Interim Staff Guidance 11, Rev. 3
3. NRC Memorandum, C. Brown to M.W. Hodges, January 29, 2004

B 3.2 ~~Deleted~~*SFSC Radiation Protection*

B 3.2.1 Deleted

HI-STORM 100 SYSTEM FSAR

APPENDIX 12.B

~~COMMENT RESOLUTION LETTERS~~

~~(XX Pages Including this Page)~~*INTENTIONALLY DELETED*

SUPPLEMENT 12.I

OPERATING CONTROLS AND LIMITS FOR THE HI-STORM 100U SYSTEM

12.I.0 INTRODUCTION

Operation of the HI-STORM 100 System with the underground HI-STORM 100U overpack is identical to operation of the HI-STORM 100 System with the aboveground overpacks up to the point where MPC transfer from the HI-TRAC transfer cask takes place. The loaded MPC is stored in the vertical orientation in the HI-STORM 100U for long-term storage operations. Operating controls and limits described in FSAR Chapter 12 are implemented for the HI-STORM 100U to the extent they are applicable. This supplement describes the unique operating controls and limits, including training requirements and technical specifications, required for the HI-STORM 100U.

12.I.1 OPERATING CONTROLS AND LIMITS

Like the aboveground overpacks, the HI-STORM 100U overpack is completely passive in performing its design functions. It provides for long-term interim storage of the spent fuel without any modifications to the MPC or HI-TRAC transfer cask designs used with the aboveground overpacks. The operating controls and limits established for the MPC and HI-TRAC transfer cask are not altered for use with the HI-STORM 100U.

Operating controls and limits pertaining to movement of the overpack do not apply to the HI-STORM 100U because it is an integral component of the ISFSI and is not able to be moved once installed. It is not required to establish a lift height limit or to evaluate a drop event because the HI-STORM 100U cannot be lifted. It is not required to measure dose rates on the side of the HI-STORM 100U because it is located underground and is, therefore, inaccessible by plant personnel and because side dose rates are limited by shielding from the surrounding soil and provide a negligible contribution to the dose rate at the controlled area boundary.

12.I.2 TRAINING MODULES

The classroom and dry run training programs described in FSAR Subsections 12.2.1 and 12.2.2 must be modified to address the differences between the use of an aboveground, moveable overpack and the belowground, immovable HI-STORM 100U. Specifically, procedures and dry runs must be created or modified to address the operational differences in using the HI-TRAC transfer cask and mating device with the HI-STORM 100U versus the aboveground overpack. These procedures and dry run training shall be based upon the operations of the HI-STORM 100U described in Supplement 8.I.

12.I.3 TECHNICAL SPECIFICATIONS

No new Limiting Conditions for Operation (LCOs) are required for use of the HI-STORM 100U. LCO 3.1.2, which requires periodic surveillance to ensure operability of the cask heat removal system, can be implemented with HI-STORM 100U more easily than for the aboveground overpack because the outlet air ducts are located at approximately ground level.

13.61 REFERENCES

[13.0.1] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[13.0.2] Holtec International Quality Assurance Program, Revision 13.

[13.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

[13.0.4] NRC QA Program Approval for Radioactive Material Packages No. 0784, Revision 3, Docket 71-0784.

SUPPLEMENT 13.I

QUALITY ASSURANCE PROGRAM FOR THE HI-STORM 100U SYSTEM

The quality assurance program described in Chapter 13 is implemented for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the HI-STORM 100U System designated as important-to-safety.