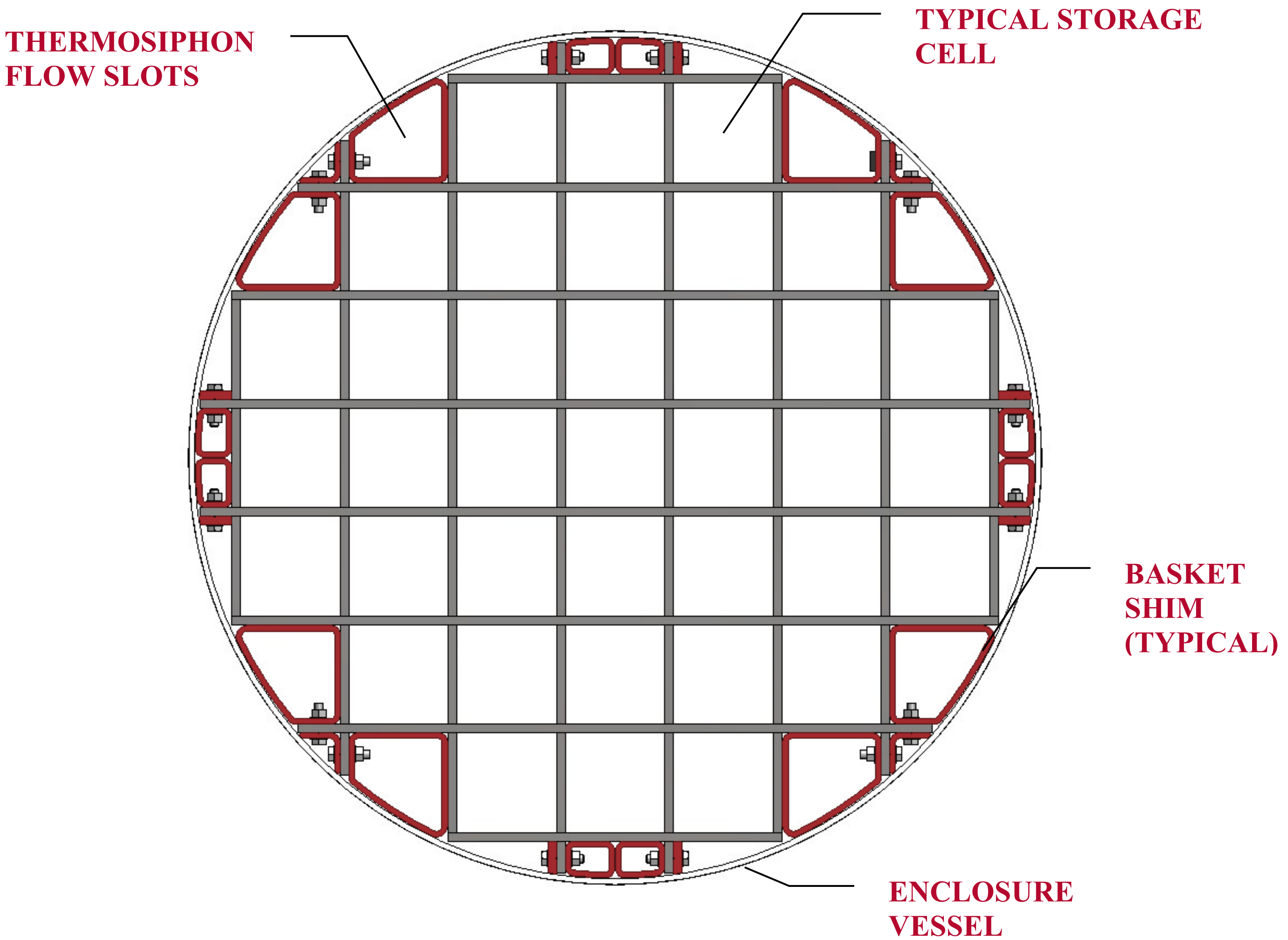
TABLE 1.0.1

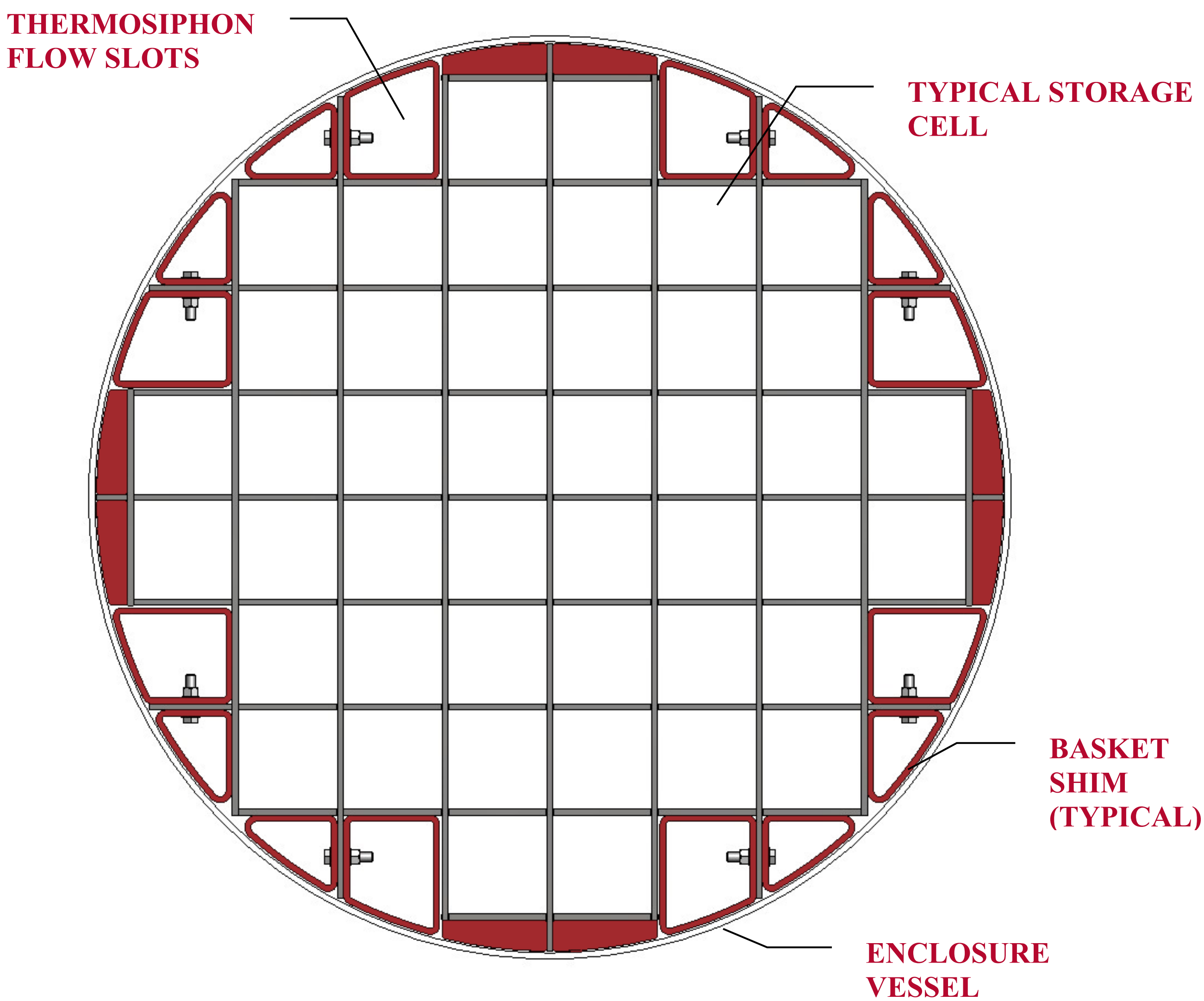
HI-STORM FW SYSTEM COMPONENTS

Item	Designation (Model Number)
Overpack	HI-STORM FW (Includes Standard, Version XL, & Version E)
PWR Multi-Purpose Canister	MPC-37, MPC-32ML, MPC-37P*, MPC-44
BWR Multi-Purpose Canister	MPC-89 (Includes Standard & Version CBS)
Transfer Cask	HI-TRAC VW(Standard), HI-TRAC VW Version V, HI-TRAC VW Version V2

*-MPC-37P qualified for storage in the HI-STORM FW Version E.



1.1.4C MPC-37P IN CROSS SECTION



1.1.4D MPC-44 IN CROSS SECTION

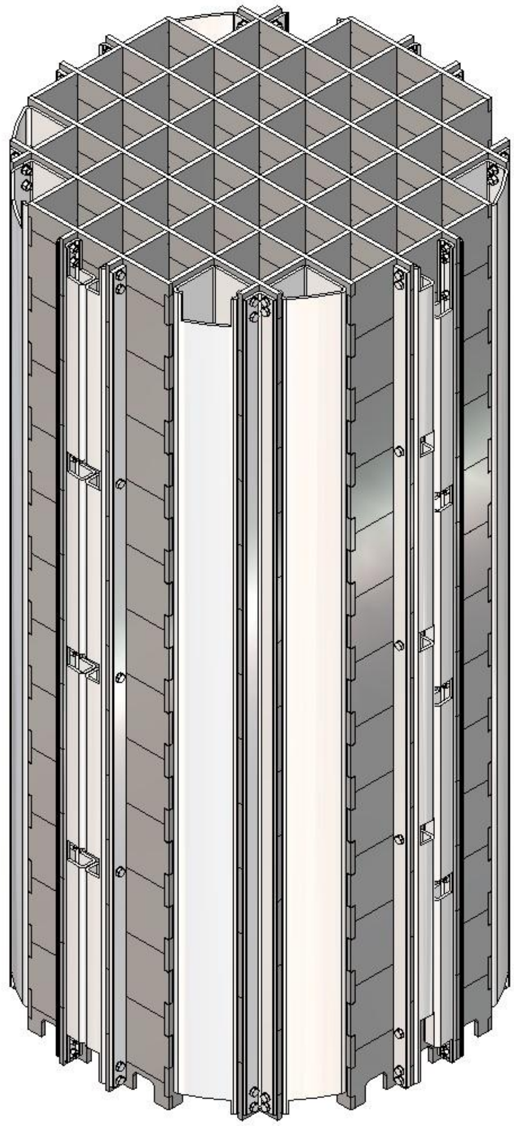


FIGURE 1.1.6C: MPC-37P PWR FUEL BASKET (37 STORAGE CELLS) WITH FUEL BASKET SHIMS IN PERSPECTIVE VIEW

FIGURE 1.1.6D: MPC-44 PWR FUEL BASKET (44 STORAGE CELLS) WITH FUEL BASKET SHIMS IN PERSPECTIVE VIEW

The MPC incorporates a redundant closure system. The MPC lid is edge-welded (welds are depicted in the licensing drawing in Section 1.5) to the MPC outer shell. The lid is equipped with vent and drain ports that are utilized to remove moisture from the MPC and backfill the MPC with a specified amount of inert gas (helium). The vent and drain ports (fabricated with or without the redundant port cover design) are closed tight and covered with a port cover (plate) that is seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid; it covers the MPC lid-to shell weld and the vent and drain port cover plates. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the suitably sized threaded anchor locations (TALs) in the MPC lid.

As discussed later in this section, the height of the MPC cavity plays a direct role in setting the amount of shielding available in the transfer cask. To maximize shielding and achieve ALARA within the constraints of a nuclear plant (such as crane capacity), it is necessary to minimize the cavity height of the MPC to the length of the fuel to be stored in it. Accordingly, the height of the MPC cavity is customized for each fuel type listed in Section 2.1. Table 3.2.1 provides the data to set the MPC cavity length as a small adder to the nominal fuel length (with any applicable NFH) to account for manufacturing tolerance, irradiation growth and thermal expansion effects.

For fuel assemblies that are shorter than the MPC cavity length (such as those without a control element in PWR SNF), a fuel shim may be utilized (as appropriate) between the fuel assembly and the MPC cavity to reduce the axial gap as prescribed by Table 3.2.1. A small axial clearance is provided to account for manufacturing tolerances and the irradiation and thermal growth of the fuel assemblies. The actual length of fuel shims (if required) will be determined on a site-specific and fuel assembly-specific basis.

All components of the MPC assembly that may come into contact with spent fuel pool water or the ambient environment are made from stainless steel alloy or aluminum/aluminum alloy materials. Prominent among the aluminum based materials used in the MPC is the Metamic–HT neutron absorber lattice that comprises the fuel basket. As discussed in Chapter 8, concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the HI-STORM FW MPCs. All structural components in an MPC enclosure vessel shall be made of Alloy X, a designation whose origin, as explained in the HI-STORM 100 FSAR [1.1.3], lies in the U.S. DOE's repository program.

As explained in Appendix 1.A, Alloy X (as defined in this FSAR) may be one of the following materials.

- Type 316
- Type 316LN
- Type 304
- Type 304LN
- Duplex Stainless Alloy S31803

Any stainless steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed above.

As discussed in Chapter 3, the licensed basis of HI-TRAC VW design is ALARA focused with the thickness of lead specified as a variable that can be optimized to maximize the cask's shielding effectiveness within the constraint of the plant's crane capacity. Therefore, it is necessary to perform the safety analysis of the HI-TRAC VW design customized for a specific site to ensure that the occupational dose is ALARA. Because the transfer cask serves no criticality or confinement function, the safety analyses pertain to structural, thermal-hydraulic and shielding compliance. Table 1.2.11 contains a list of all safety evaluations that must be performed on a HI-TRAC VW cask embodiment to qualify it for use at a plant site. As can be seen from Table 1.2.11, the required evaluations must be performed for the specific site conditions with the provision that the analysis methodology does not violate that documented in this FSAR.

HI-TRAC VW is principally made of carbon steel and lead. The cask consists of two major parts, namely (a) a multi-shell cylindrical cask body, and (b) a quick connect/disconnect bottom lid. The cylindrical cask body is made of three concentric shells joined to a solid annular top flange and a solid annular bottom flange by circumferential welds. The innermost and the middle shell are fixed in place by longitudinal ribs which serve as radial connectors between the two shells. The radial connectors provide a continuous path for radial heat transfer and render the dual shell configuration into a stiff beam under flexural loadings. The space between these two shells is occupied by lead, which provides the bulk of the transfer cask's gamma radiation shielding capability and accounts for a major portion of its weight.

Between the middle shell and the outermost shell is the weldment that is referred to as the "water jacket." The water jacket is filled with water and may contain ethylene glycol fortified water, if warranted by the environmental conditions at the time of use. Addition of ethylene glycol is not required due to environmental conditions if the MPC heat load is high enough to preclude freezing of the water. This threshold MPC heat load is determined on a site-specific basis through the methodology described in Chapter 4. The water jacket provides most of the neutron shielding capability to the cask. The water jacket is outfitted with pressure relief devices to prevent over-pressurization in the case of an off-normal or accident event that causes the water mass inside of it to boil.

The water in the water jacket serves as the neutron shield when required. When the cask is being removed from the pool and the MPC is full of water, the water jacket can be empty. This will minimize weight, if for example, crane capacities are limited, since the water within the MPC cavity is providing the neutron shielding during this time. However, the water jacket must be filled before the MPC is emptied of water. This keeps the load on the crane (i.e., weight of the loaded transfer cask) nearly constant between the lifts before and after MPC processing. Furthermore, the amount of shielding provided by the transfer cask is maximized at all times within crane capacity constraints. The water jacket concept is disclosed in a Holtec Patent [6,587,536 B1].

As the description of loading operations in Chapter 9 of this FSAR indicates, most of the human activities occur near the top of the transfer cask. Therefore, the geometry of the transfer cask is configured to minimize the use of penetrations and discontinuities and maximize shielding in areas where penetrations and discontinuities are necessary. The standard version HI-TRAC VW is lifted using a pair of lift blocks that are anchored into the top forging of the transfer cask using a set of

- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

Other aspects of a shielding material, such as ease of handling and prior nuclear industry use, are also considered. Final specification of a shield material is a result of optimizing the material properties with respect to the above criteria, along with the design of the shield system, to achieve the desired shielding results.

The HI-TRAC VW transfer cask is equipped with a water jacket or neutron shield cylinder (NSC) to provide radial neutron shielding. The water in the water jacket may be fortified with ethylene glycol to prevent freezing under low temperature operations [1.2.4], depending on MPC heat loads. The NSC neutron shielding is provided by Holtite-A. Information on Holtite-A is provided in Appendix 1.B of the HI-STORM 100 FSAR [1.1.3].

Neutron shielding in the HI-TRAC VW Version V2 transfer cask in the radial direction is provided by Holtite-A within the removable NSC. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A contains a nominal B_4C loading as specified in Table 1.2.5. Appendix 1.B of HI-STORM 100 System FSAR [1.1.3] provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

In the following, a brief summary of the performance characteristics and properties of Holtite-A is provided.

Density

The specific gravity of Holtite-A is specified in Table 1.2.5 and Appendix 1.B of [1.1.3]. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4%. The density used for the shielding analysis is specified in Table 1.2.5 and is conservatively assumed to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The weight concentration of hydrogen is specified in Table 1.2.5.

Boron Carbide

Boron carbide is dispersed within Holtite-A in finely dispersed powder form. For the HI-STORM FW System, Holtite-A is specified with a nominal B_4C weight percent listed in Table 1.2.5

Design Temperature

The design temperatures of Holtite-A are provided in Table 1.B.1. of [1.1.3]. The maximum spatial temperatures of Holtite-A under all normal operating conditions must be demonstrated to be below

1.2.1.7 HI-DRIP Auxiliary Cooling System

The HI-DRIP auxiliary cooling system is designed to prevent the water in the loaded MPC from boiling during the interval after it has been lifted out of the fuel pool and prior to the evacuation of water and backfill of helium. Figure 1.2.10 provides an illustration of the HI-DRIP system.

HI-DRIP is an optional ancillary that consists of a tubular ring that girdles the transfer cask above its main cylindrical body. The ring is equipped with small spray nozzles uniformly spaced around its circumference such that the entire circumference of the transfer cask shell can be drenched by the spray from them. The ring is connected to the plant's water supply with a gate valve serving to regulate the flow of water to the ring. The amount of water sprayed should be set such that the entire surface of the cask is wetted by gravity. While some of the water will evaporate, the residual water may be collected in a trough at the bottom of the cask equipped with a provision to drain it out to either the pool or the plant's drainage system. As long as the external surface of the transfer cask remains wet, the water in the canister will remain below boiling. It is noted that the operation of HI-DRIP does not require any pump or electric power; the motive pressure is provided entirely by the plant's water supply system.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The design features of the HI-STORM FW System, described in Subsection 1.2.1 in the foregoing, are intended to meet the following principal performance characteristics under all credible modes of operation:

(a) Maintain subcriticality
(b) Prevent unacceptable release of contained radioactive material
(c) Minimize occupational and site boundary dose
(d) Permit retrievability of contents (fuel must be retrievable under normal and off-normal conditions in accordance with ISG-2 [1.2.18] and the MPC must be recoverable after accident conditions in accordance with ISG-3)

Chapter 11 identifies the many design features built into the HI-STORM FW System to minimize dose and maximize personnel safety. Among the design features intrinsic to the system that facilitate meeting the above objectives are:

i. The loaded HI-STORM FW overpack and loaded HI-TRAC VW transfer cask are typically maintained in a vertical orientation during handling (except as described in Subsection 4.5.1).

ii. The height of the HI-STORM FW overpack and HI-TRAC VW transfer cask is minimized consistent with the length of the SNF. This eliminates the need for major structural modifications at the plant and/or eliminates operational steps that impact ALARA.

1.2.3 Cask Contents

This sub-section contains information on the cask contents pursuant to 10 CFR72, paragraphs 72.2(a)(1),(b) and 72.236(a),(c),(h),(m).

The HI-STORM FW System is designed to house both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key system data and parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the Glossary. All fuel assemblies, non-fuel hardware, and neutron sources authorized for packaging in the MPCs must meet the fuel specifications provided in Section 2.1. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers (DFC) or fuel cell storage location equipped with a damaged fuel isolator (DFI) for damaged fuel that can be handled by normal means. Figure 2.1.7 shows a typical DFI.

As shown in Figure 1.2.1a (MPC-37) and Figure 1.2.2 (MPC-89), each storage location is assigned to one of three regions, denoted as Region 1, Region 2, and Region 3 with an associated cell identification number. For example, cell identified as 2-4 is Cell 4 in Region 2. A damaged fuel assembly in a DFC or using a DFI can be stored in the outer peripheral locations of the MPC-37/MPC-32ML/MPC-37P/MPC-44 and MPC-89 as shown in Figures 2.1.1 and 2.1.2, respectively. The permissible heat loads for each cell, region, and the total canister are given in Tables 1.2.3 and 1.2.4 for the MPC-37/MPC-32ML/MPC-37P/MPC-44 and MPC-89, respectively. The sub-design heat loads for each cell, region and total canister are in Table 4.4.11.

As an alternative to the loading patterns discussed above, fuel storage in the MPC-37 and MPC-89 is permitted to use the heat load patterns shown in Figure 1.2.3 through Figure 1.2.5 (MPC-37) and Figures 1.2.6 and 1.2.7 (MPC-89).

A minor deviation from the prescribed loading pattern in an MPC's permissible contents to allow one slightly thermally-discrepant fuel assembly per quadrant to be loaded as long as the peak cladding temperature for the MPC remains below the ISG-11 Rev 3 requirements is permitted for essential dry storage campaigns to support decommissioning.

TABLE 1.2.1†

KEY SYSTEM DATA FOR HI-STORM FW SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs‡	5~~3~~	4~~2~~ for PWR 1 for BWR
MPC storage capacity:	MPC-37	Up to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware, of classes specified in Table 2.1.1a. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1a with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37. Alternative damaged fuel patterns are shown in Figures 1.2.3 through 1.2.5.
MPC storage capacity:	MPC-89	Up to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 89. Alternative damaged fuel patterns are shown in Figure 1.2.6 and 1.2.7.
MPC storage capacity:	MPC-32ML	Up to 32 undamaged ZR clad PWR fuel assemblies, of classes specified in Table 2.1.1b. Up to 8 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1b with the remaining basket cells containing undamaged fuel assemblies, up to a total of 32.
MPC storage capacity:	MPC-37P	Up to 37 undamaged ZR clad PWR fuel assemblies, of classes specified in Table 2.1.1c. Up to

† Damaged fuel assemblies which can be handled by normal means can be stored in the designated locations for damaged fuel using DFIs or DFCs

‡ See Chapter 2 for a complete description of authorized cask contents and fuel specifications.

TABLE 1.2.1†

KEY SYSTEM DATA FOR HI-STORM FW SYSTEM

ITEM	QUANTITY	NOTES
		12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1c with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37.
MPC storage capacity:	MPC-44	Up to 44 undamaged ZR clad PWR fuel assemblies, of classes specified in Table 2.1.1d. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1d with the remaining basket cells containing undamaged fuel assemblies, up to a total of 44.

TABLE 1.2.2

KEY PARAMETERS FOR HI-STORM FW MULTI-PURPOSE CANISTERS

Parameter	PWR	BWR
Pre-disposal service life (years)	100	100
Design temperature, max./min. (˚F)	752†/-40††	752†/-40††
Design internal pressure (psig) Normal conditions Off-normal conditions Accident Conditions	 100 120 200	 100 120 200
Total heat load, max. (kW)	See Table 1.2.3a/b/c/e	See Table 1.2.4
Maximum permissible peak fuel cladding temperature: Long Term Normal (˚F) Short Term Operations (˚F) Off-normal and Accident (˚F)	 752 752 or 1058††† 1058	 752 752 or 1058††† 1058
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
B_4C content (by weight) (min.) in the Metamic-HT Neutron Absorber (storage cell walls)	10%	10%
PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10CFR2.390		
PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10CFR2.390		
End closure(s)	Welded	Welded
Fuel handling	Basket cell openings compatible with standard grapples	Basket cell openings compatible with standard grapples
Heat dissipation	Passive	Passive

† Maximum normal condition design temperatures for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.2.3.

†† Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2 and no fuel decay heat load.

††† See Section 4.5 for discussion of the applicability of the 1058°F temperature limit during short-term operations, including MPC drying.

TABLE 1.2.3a MPC-37 and MPC-37P HEAT LOAD DATA (See Figure 1.2.1a for regions)					
Number of Regions: 3					
Number of Storage Cells: 37					
Maximum Design Basis Heat Load (kW): 44.09 (Pattern A); 45.0 (Pattern B) (Note 3)					
Region No.	Decay Heat Limit per Cell, kW (Notes 1 and 2)		Number of Cells per Region	Decay Heat Limit per Region, kW	
	Pattern A	Pattern B		Pattern A	Pattern B
1	1.05	1.0	9	9.45	9.0
2	1.70	1.2	12	20.4	14.4
3	0.89	1.35	16	14.24	21.6

Notes:

(1) See Chapter 4 for decay heat limits per cell when vacuum drying high burnup fuel.

(2) Decay heat limit per cell for cells containing damaged fuel or fuel debris is equal to the decay heat limit per cell of the region where the damaged fuel or fuel debris is permitted to be stored.

(3) Alternative heat load patterns for the MPC-37 are included in Table 1.2.3d.

(4) Alternative heat load pattern for the MPC-37P is included in Table 1.2.3c.

TABLE 1.2.3c Alternative MPC-37P HEAT LOAD DATA (see Figure 1.2.1c)	
Number of Storage Cells:	37
Maximum Design Basis Heat Load (kW):	45
Maximum Quadrant Heat Load (kW):	11.25
Decay Heat Limit per Cell (kW):	See Figures 1.2.9a/b

Notes:

(1) See Chapter 4 for decay heat limits per cell when vacuum drying moderate or high burnup fuel.

(2) Decay heat limit per cell for cells containing damaged fuel or fuel debris is equal to the decay heat limit per cell of the region where the damaged fuel or fuel debris is permitted to be stored.

TABLE 1.2.3e MPC-44 HEAT LOAD DATA (See Figure 1.2.1d)	
Number of Regions:	1
Number of Storage Cells:	44
Maximum Design Basis Heat Load (kW):	44
Decay Heat Limit per Cell, kW:	1.0

Notes:

(1) See Chapter 4 for decay heat limits per cell when vacuum drying moderate or high burnup fuel.

(2) There is a 5% decay heat penalty per cell for cells permitted to contain damaged fuel or fuel debris.

TABLE 1.2.5
CRITICALITY AND SHIELDING SIGNIFICANT SYSTEM DATA

Item	Property	Value
Metamic-HT Neutron Absorber	Nominal Thickness (mm)	10 (MPC-89) 15 (MPC-37) 15 (MPC-32ML) 20 (MPC-37P) 13 (MPC-44)
	Minimum B_4C Weight %	10 (MPC-89) 10 (MPC-37) 10 (MPC-32ML) 10 (MPC-37P) 10 (MPC-44)
Concrete in HI-STORM FW standard overpack body and Domed lid	Installed Nominal Density (lb/ft^3)	150 (reference) 250 (maximum)
Concrete in HI-STORM FW Standard lid and Version XL lid	Installed Nominal Density (lb/ft^3)	150 (reference) 200 (maximum)
Concrete in HI-STORM FW Version E overpack body	Installed Nominal Density (lb/ft^3)	175 (reference) 250 (maximum)
Concrete in HI-STORM FW Version E lid	Installed Nominal Density (lb/ft^3)	175 (reference) 200 (maximum)
Holtite - A	**PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10CFR2.390**	

TABLE 1.2.8b

PROPRIETARY INFORMATION WITHHELD IN ACCORDANCE WITH 10CFR2.390

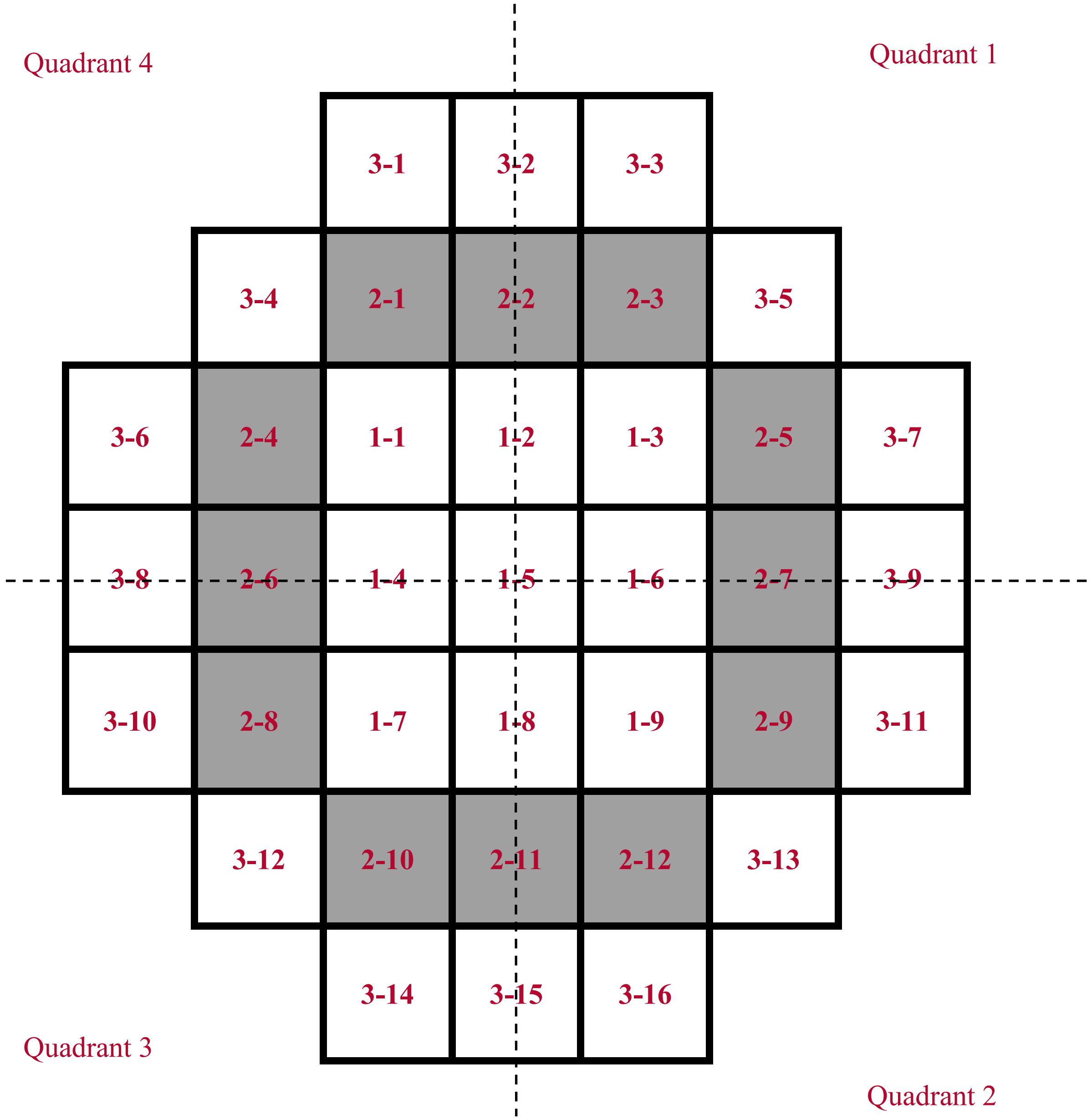


Figure 1.2.1c: MPC-37P Basket Cell Identification

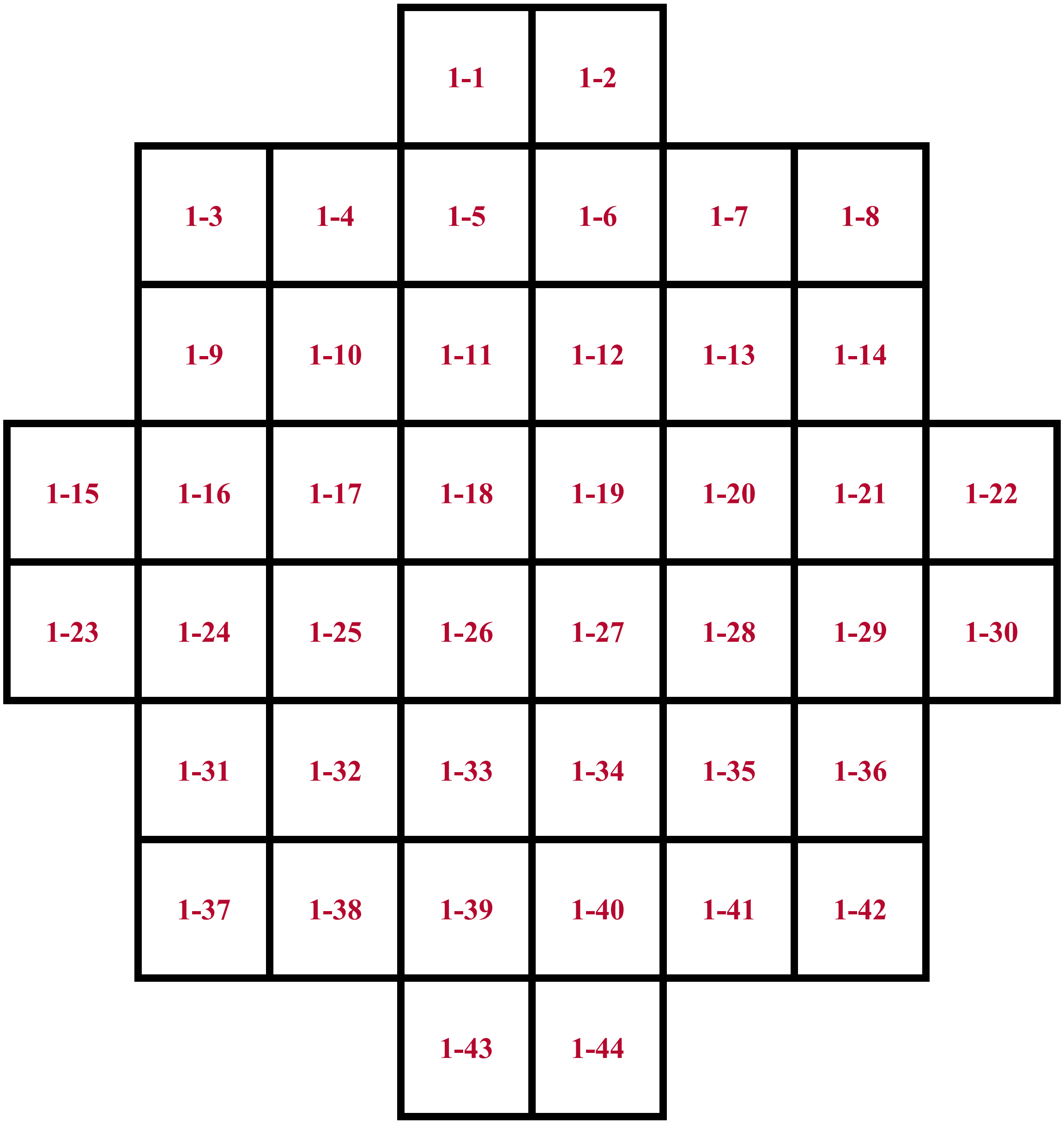


Figure 1.2.1d: MPC-44 Basket Cell Identification

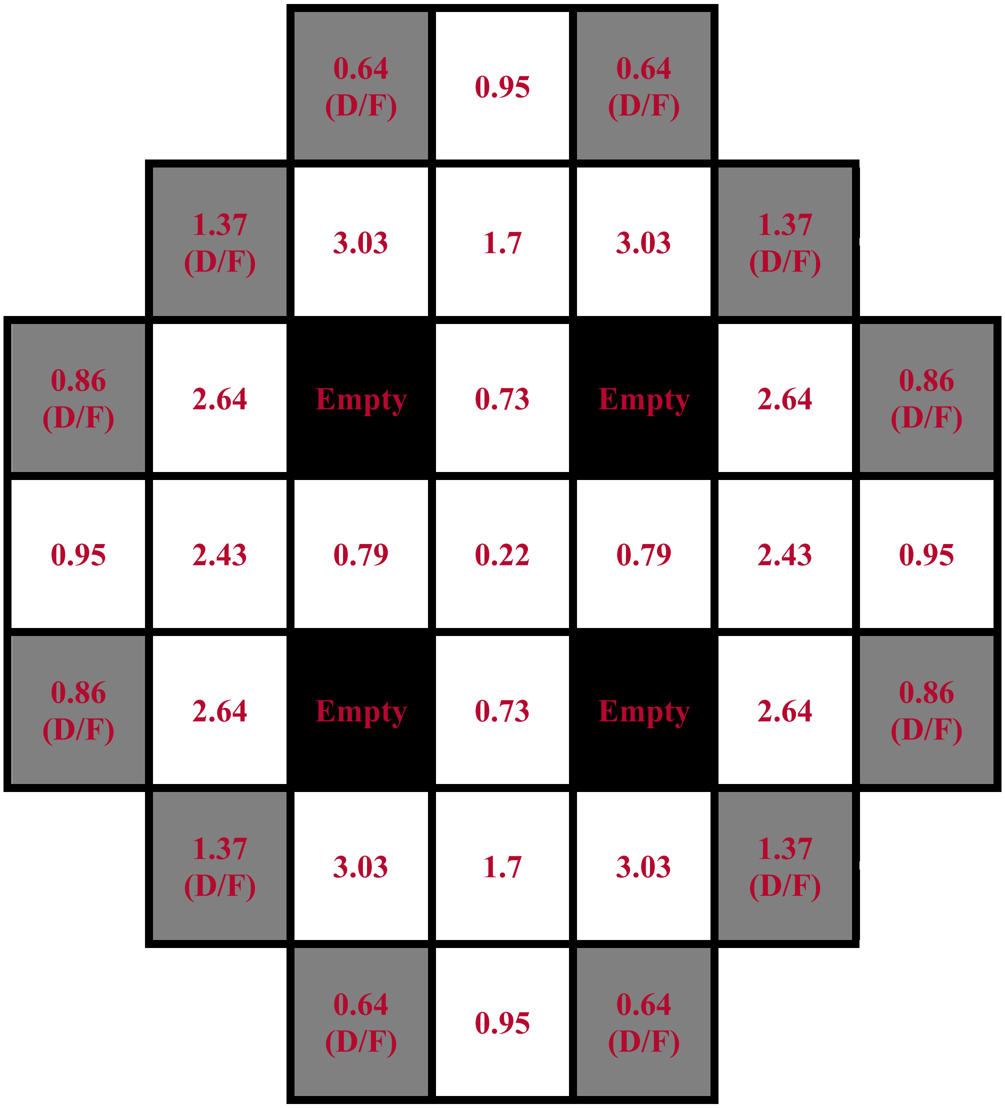


Figure 1.2.9a:
Loading Pattern 1 for MPC-37P

(**All Storage cell heat loads are in kW,** Undamaged Fuel, or Damaged Fuel in DFCs and/or using DFIs, and/or Fuel Debris in a DFC may be stored in cells denoted by "D/F." Cells denoted as "Empty" must remain empty regardless of the contents of the adjacent cell)

		0.64	0.95	0.64		
	1.37	3.03 (D/F)	1.7	3.03 (D/F)	1.37	
0.86	2.64	Empty	0.73	Empty	2.64	0.86
0.95	2.43	0.79	0.22	0.79	2.43	0.95
0.86	2.64	Empty	0.73	Empty	2.64	0.86
	1.37	3.03 (D/F)	1.7	3.03 (D/F)	1.37	
		0.64	0.95	0.64		

Figure 1.2.9b:
Loading Pattern 2 for MPC-37P

(**All Storage cell heat loads are in kW,** Undamaged Fuel, or Damaged Fuel in DFCs and/or using DFIs, and/or Fuel Debris in a DFC may be stored in cells denoted by “D/F.” Cells denoted as “Empty” must remain empty regardless of the contents of the adjacent cell)

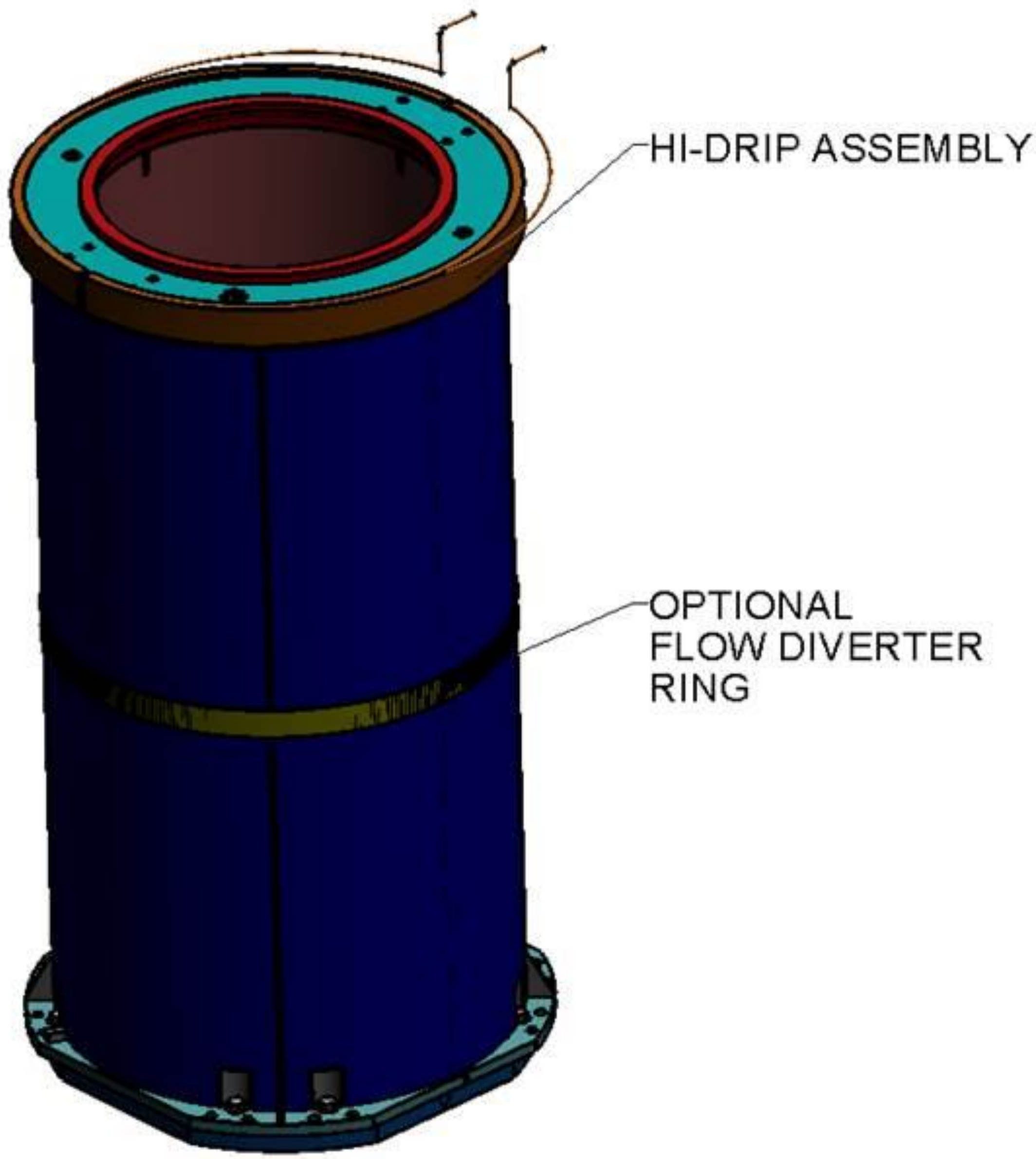


Figure 1.2.10: HI-DRIP AUXILIARY COOLING SYSTEM

modification, and decommissioning of structures, systems, and components important to safety is incorporated by reference into this FSAR. Holtec International's QA program has been certified by the USNRC (Certificate No. 71-0784).

The HI-STORM FW System will be fabricated by Holtec International Manufacturing Division (HMD) located in Pittsburgh, Pennsylvania and Camden, New Jersey. HMD is a long term N-Stamp holder and fabricator of nuclear components. In particular, HMD has been manufacturing HI-STORM and HI-STAR system components since the inception of Holtec International's dry storage and transportation program in the 1990s. HMD routinely manufactures ASME code components for use in the US and overseas nuclear plants. Both Holtec International's headquarters and the HMD subsidiary have been subject to triennial inspections by the USNRC. If another fabricator is to be used for the fabrication of any part of the HI-STORM FW System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

The Metamic-HT is fabricated by Holtec affiliate, Orrvilon located in Orrville, Ohio. Orrvilon's QA program is controlled by Holtec International. If another fabricator is to be used for the fabrication of Metamic-HT, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

Holtec International's Nuclear Power Division (NPD) also carries out site services for dry storage deployments at nuclear power plants. Several nuclear plants, such as Trojan (completed) and Waterford (ongoing, ca. 2009) have deployed dry storage at their sites using a turn key contract with Holtec International.

1.5 DRAWINGS

The following HI-STORM FW System drawings are provided on subsequent pages in this section to fulfill the requirements in 10 CFR 72.2(a)(1),(b) and 72.230(a):

Drawing No.	Title	Revision
6494	HI-STORM FW BODY	23
6508	HI-STORM FW STANDARD LID ASSEMBLY	8
6514	HI-TRAC VW – MPC-37	14
6799	HI-TRAC VW – MPC-89	11
11006	HI-TRAC VW Version V	0
11283	HI-TRAC VW Version V2	2
10115	HI-TRAC VW Version P – MPC-89	1
6505	MPC-37 ENCLOSURE VESSEL	26
6506	MPC-37 FUEL BASKET	15
6512	MPC-89 ENCLOSURE VESSEL	28
6507	MPC-89 FUEL BASKET	15
10464	MPC-32ML ENCLOSURE VESSEL	1
10457	MPC-32ML FUEL BASKET	0
9964	HI-STORM FW Version XL Lid Assembly	3
10455	HI-STORM FW Domed Closure Lid	6
11501	HI-STORM FW Version E Lid Assembly	2
11621	HI-STORM FW Version E Overpack	1
12030	MPC-89 Version CBS Fuel Basket	0
12283	MPC-37P FUEL BASKET	0
12288	MPC-44 FUEL BASKET	0

Critical dimensions for the HI-STORM FW Version E Overpack are recorded in report [1.5.1].

Notes:

1. The HI-TRAC VW for MPC-37 is the designated HI-TRAC for all PWR MPCs (MPC-37, ~~and~~ MPC-32ML, MPC-37P, and MPC-44).
2. The Enclosure Vessel for MPC-37 is the designated Enclosure Vessel for MPC-37P and MPC-44.

PROPRIETARY DRAWINGS WITHHELD IN ACCORDANCE WITH 10CFR2.390

certain necessary alternatives, as discussed in Section 2.2. The principal exception to the above Code pertains to the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell, as discussed in Section 2.2. In addition, Threaded Anchor Locations (TALs) in the MPC lid are designed in accordance with the requirements of NUREG-0612 for critical lifts to facilitate handling of the loaded MPC.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid-to-shell weld is further ensured by performing a progressive liquid penetrant examination of the weld layers, and a Code pressure test.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, pressure testing, and helium leak testing (helium leak testing not required for redundant port cover design) provides assurance of canister closure integrity in lieu of the specific weld joint configuration requirements of Section III, Subsection NB.

Compliance with the ASME Code, with respect to the design and fabrication of the MPC, and the associated justification are discussed in Section 2.2. The MPC design is analyzed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. The required characteristics of the fuel assemblies to be stored in the MPC are limited in accordance with Section 2.1.

Thermal

The thermal design and operation of the MPC in the HI-STORM FW System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.1]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

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

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

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

iv. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65ºC (117ºF) and the number of excursions to less than 10.

To achieve compliance with the above criteria, certain design and operational changes are necessary, as summarized below.

i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400ºC (752ºF). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570ºC (1058ºF) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.2]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions have been added to ensure these limits are met.

ii. A method of drying, such as forced helium dehydration (FHD) is used if the above temperature limits for short-term operations cannot be met.

iii. The off-normal and accident condition PCT limit remains unchanged at 570 ºC (1058ºF).

The MPC cavity is dried, either with FHD or vacuum drying (continuous or cyclic), and then it is backfilled with high purity helium to promote heat transfer and prevent cladding degradation.

The normal condition design temperatures for the stainless steel components in the MPC are provided in Table 2.2.3.

The MPC-37 and MPC-89 models allow for regionalized storage where the basket is segregated into three regions as shown in Figures 1.2.1a and 1.2.2. Decay heat limits for regionalized loading are presented in Tables 1.2.3a and 1.2.4 for MPC-37 and MPC-89, respectively. MPC-37P follows a storage pattern shown in Figure 1.2.9, while MPC-44 is uniformly loaded as specified in Table 1.2.3.c. Specific requirements, such as approved locations for DFCs, DFIs, and non-fuel hardware are given in Section 2.1.

As an alternative to the regionalized storage patterns, The MPC-37 and MPC-89 models allow for the use of the heat load charts shown in Figures 1.2.3 through 1.2.5 (MPC-37) and 1.2.6 through 1.2.7 (MPC-89).

Shielding

The dose limits for an ISFSI using the HI-STORM FW System are delineated in 10CFR72.104 and 72.106. Compliance with these regulations for any particular array of casks at an ISFSI is necessarily site-specific and must be demonstrated by the licensee. Dose for a single cask and a representative cask array is illustrated in Chapter 5.

The MPC provides axial shielding at the top and bottom ends to maintain occupational exposures ALARA during canister closure and handling operations. The HI-TRAC VW bottom lid also contains shielding. The occupational doses are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 9).

evaluation of the potential for brittle fracture in structural steel materials is presented in Section 3.1.

The HI-TRAC VW is designed and evaluated for the maximum heat load analyzed for storage operations. The maximum allowable temperature of water in the HI-TRAC jacket is a function of the internal pressure. To preclude over- pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is restricted to be less than the saturation temperature at the shell design pressure. Even though the analysis shows that the water jacket will not over-pressurize, a relief device is placed at the top of the water jacket shell. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the minimum allowable operating temperature and by either adding ethylene glycol or applying a minimum heat load restriction when loading the MPC. To preclude over-pressurization of the HI-TRAC VW Version V2 neutron shield cylinder (NSC) during a fire accident, the NSC is fitted with a relief device. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1. The working area ambient temperature limit for loading operations is limited in accordance with Table 2.2.2.

Shielding

The HI-TRAC VW transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10CFR20, while also maintaining the maximum load on the plant's crane hook to below the rated capacity of the crane. The HI-TRAC VW Version V2 design includes a detachable NSC that can be removed for movements of the MPC and HI-TRAC into and out of the pool such that the amount of gamma shielding is maximized relative to the crane capacity while the water in the MPC provides neutron shielding. The HI-TRAC is then placed inside of and connected to the NSC such that it provides neutron shielding after the water is drained from the MPC. As discussed in Subsection 1.2.1, the shielding in HI-TRAC VW is maximized within the constraint of the allowable weight at a plant site. The HI-TRAC VW calculated dose rates for a set of reference conditions are reported in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations, as described in Chapter 11. A postulated HI-TRAC VW accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Chapter 5.

The annular area between the MPC outer surface and the HI-TRAC VW inner surface can be isolated to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC VW surfaces expected to require decontamination are coated with a suitable coating. The maximum permissible surface contamination for the HI-TRAC VW is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 11).

Confinement

The HI-TRAC VW transfer cask does not perform any confinement function. The HI-TRAC VW provides physical protection and radiation shielding of the MPC contents during MPC loading, unloading, and transfer operations.

Table 2.0.13 – HI-TRAC VW Version P – MPC-89 (Drawing # 10115)		
Item Number	Part Name	ITS QA Safety Category
Per Licensing Drawing in Section 1.5		

Table 2.0.14 – MPC-37P (Drawing # 12283)		
Item Number	Part Name	ITS QA Safety Category
Per Licensing Drawing in Section 1.5		

Table 2.0.15 – MPC-44 (Drawing # 12288)		
Item Number	Part Name	ITS QA Safety Category
Per Licensing Drawing in Section 1.5		

geometrically constrained to prevent their ejection from the storage cavity during a postulated accident event.

4. The design configuration of the DFI is common for all Light Water Reactor fuel.
5. The DFI is constructed entirely from stainless steel or nickel alloy suitable for use within the high temperature environment of the MPC. See Table 2.1.11 for critical characteristics of the DFI. Per Figure 2.1.7, the cap walls shall have perforation with a maximum size as listed in Table 2.1.11. This allows flow through the walls while keeping gross particulate fissile material inside the basket cell.
6. The DFI includes perforated plates at the top and bottom to allow for flow through the basket cell while keeping gross particulate fissile material inside the basket cell.
7. The load bearing members of the DFI will be designed to satisfy Level D Stress limits per ASME Section III, Appendix F.

2.1.4 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, cross sectional dimensions, and weight. These parameters, which define the mechanical and structural design, are specified in Subsection 2.1.8. An appropriate axial clearance is provided to prevent interference due to the irradiation and thermal growth of the fuel assemblies.

2.1.5 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the fuel's peak cladding temperature (PCT) which is a function of the maximum decay heat per assembly and the decay heat removal capabilities of the HI-STORM FW System.

To ensure the permissible PCT limits are not exceeded, Subsection 1.2 specifies the maximum allowable decay heat per assembly for each MPC model ~~in the three region configuration~~ (see also Tables 1.2.3 and 1.2.4).

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The design basis fuel assembly for thermal calculations for both PWR and BWR fuel is provided in Table 2.1.4.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in references [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.5 and Figures 2.1.3 and 2.1.4. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM FW System.

Minimum cooling time must also meet limits specified in Tables 2.1.1a and 2.1.1b. If the calculated *Ct* is less than the cooling time limits in Tables 2.1.1a or 2.1.1b, the minimum cooling time in table is used.

For MPC-37 and MPC-89, the coefficients for above equation for the assembly in an individual cell depend on the heat load limit in that cell, Table 2.1.10 lists the coefficients for several heat load limit ranges. Note that the heat load limits are only used for the lookup of the coefficients in that table, and do not imply any equivalency. Specifically, meeting heat load limits is not a substitute for meeting burnup and cooling time limits, and vice versa.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1b.

2.1.6.2 Radiological Parameters for Spent Fuel and Non-fuel Hardware in MPC-37P and MPC-44

MPC-37P is authorized to store CE15x15 spent fuel with burnup - cooling time combinations as given in Table 2.1.12. MPC-44 is authorized to store 14x14 spent fuel with burnup - cooling time combinations as given in Table 2.1.12.

The burnup and cooling time for every fuel assembly loaded into the MPC-37P and MPC-44 must satisfy the following equation:

$$Ct = A \cdot Bu^4 + B \cdot Bu^3 + C \cdot Bu^2 + D\,Bu + E$$

where,

Ct = Minimum cooling time (years),

Bu = Assembly-average burnup (MWd/mtU),

A, B, C, D, E = Polynomial coefficients listed in Table 2.1.12.

2.1.7 Criticality Parameters for Design Basis SNF

Criticality control during loading of the MPC-37 is achieved through either meeting the soluble boron limits in Table 2.1.6 OR verifying that the assemblies meet the minimum burnup requirements in Table 2.1.7. MPC-37P is bounded by MPC-37, thus criticality control for the MPC-37 (either soluble boron credit or burnup credit) is also applicable for MPC-37P. Criticality control during loading of the MPC-32ML and MPC-44 is achieved through meeting the soluble boron limits in Table 2.1.6.

For those spent fuel assemblies that need to meet the burnup requirements specified in Table 2.1.7, a burnup verification shall be performed in accordance with either Method A OR Method B described below.

Method A: Burnup Verification Through Quantitative Burnup Measurement

For each assembly in the MPC-37 where burnup credit is required, the minimum burnup is determined from the burnup requirement applicable to the loading configuration chosen for the cask (see Table 2.1.7). A measurement is then performed that confirms that the fuel assembly burnup exceeds this minimum burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. The assembly burnup value to be compared with the minimum required burnup should be the measured burnup value as adjusted by reducing the value by a combination of the uncertainties in the calibration method and the measurement itself.

Method B: Burnup Verification Through an Administrative Procedure and Qualitative Measurements

Depending on the location in the basket, assemblies loaded into a specific MPC-37 can either be fresh, or have to meet a single minimum burnup value. The assembly burnup value to be compared with the minimum required burnup should be the reactor record burnup value as adjusted by reducing the value by the uncertainties in the reactor record value. An administrative procedure shall be established that prescribes the following steps, which shall be performed for each cask loading:

- Based on a review of the reactor records, all assemblies in the spent fuel pool that have a burnup that is below the minimum required burnup of the loading curve for the cask to be loaded are identified.
- After the cask loading, but before the release for shipment of the cask, the presence and location of all those identified assemblies is verified, except for those assemblies that have been loaded as fresh assemblies into the cask.

Additionally, for all assemblies to be loaded that are required to meet a minimum burnup, a measurement shall be performed that verifies that the assembly is not a fresh assembly.

Table 2.1.1c

MATERIAL TO BE STORED

PARAMETER	**VALUE**
	MPC-37P
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the 15x15I array/class only.
Cladding Type	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6.
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 1.6 years and meeting the equation in Subsection 2.1.6 Maximum Assembly Average Burnup: 68.2 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup†	Minimum Cooling Time: 1.6 year Maximum Burnup: - WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable
Decay heat per fuel storage location	Loading per Tables 1.2.3a and 1.2.3c.
Fuel Assembly Nominal Length (in)	≤ 150 (including NFH) ≤ 160.5 (with DFC)
Fuel Assembly Width (in)	≤ 8.52 (nominal design)
Fuel Assembly Weight (lb)	≤ 1360 (without NFH) ≤ 1510 (with NFH) ≤ 1610 (including DFC and NFH).
Other Limitations	▪ Quantity is limited to 37 undamaged ZR clad PWR class 15x15Ifuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers or damaged fuel isolators containing class 15x15I PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1c with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 37. ▪ TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location.

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.1d

MATERIAL TO BE STORED

PARAMETER	**VALUE**
	MPC-44
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the 14x14A and 14x14B array/classes only.
Cladding Type	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment (w/o)	5.0
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years and meeting the equation in Subsection 2.1.6 Maximum Assembly Average Burnup: 60.0 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup†	Minimum Cooling Time: 3 years Maximum Burnup: - WABAs and vibration suppressors: 60 GWd/mtU - TPDs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable
Decay heat per fuel storage location	Uniform Loading per Table 1.2.3e.
Fuel Assembly Nominal Length (in)	≤ 159.977 ≤ 170.5 (including DFC)
Fuel Assembly Width (in)	≤ 7.81 (nominal design)
Fuel Assembly Weight (lb)	≤ 1150 ≤ 1250 (including DFC)
Other Limitations	▪ Quantity is limited to 44 undamaged ZR clad PWR classes 14x14A or 14x14B fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers or damaged fuel isolators containing classes 14x14A or 14x14B PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1d with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 44. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location.

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.3 (continued)

BWR FUEL ASSEMBLY CHARACTERISTICS (Notes 1, 17)

Fuel Assembly Array and Class	**10x10 C**	**10x10 F**	**10x10 G**	**10x10 I**	**10x10 J**	**11x11 A**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 14)	≤ 4.8	≤ 4.7 (Note 13)	≤ 4.6 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations (Note 16)	96	92/78 (Note 7)	96/84	91/79 (Note 18)	96/80 (Note 19)	112/92 (Note 20)
Fuel Clad O.D. (in.)	≥ 0.3780	≥ 0.4035	≥ 0.387	≥ 0.4047	≥ 0.3999	≥ 0.3701
Fuel Clad I.D. (in.)	≤ 0.3294	≤ 0.3570	≤ 0.340	≤ 0.3559	≤ 0.3603	≤ 0.3252
Fuel Pellet Dia. (in.)	≤ 0.3224	≤ 0.3500	≤ 0.334	≤ 0.3492	≤ 0.3501	≤ 0.3193
Fuel Rod Pitch (in.)	≤ 0.488	≤ 0.510	≤ 0.512	≤ 0.5100	≤ 0.5149	≤ 0.4705
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)	1 (Note 5)	1	1 (Note 5)
Water Rod Thickness (in.)	≥ 0.031	≥ 0.030	≥ 0.031	≥ 0.0315	≥ 0.0297	≥ 0.0340
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.060	≤ 0.100	≤ 0.0938	≤ 0.100

16. Any number of fuel rods in an assembly can be replaced by irradiated or unirradiated Steel or Zirconia rods. If the rods are irradiated, the site specific dose and dose rate analyses performed under 10 CFR 72.212 should include considerations for the presence of such rods.
17. Any number of fuel rods in an assembly can contain BLEU fuel. If the BLEU fuel rods are present, the site specific dose and dose rate analyses performed under 10 CFR 72.212 should include consideration for the presence of such rods.
18. Contains in total 91 fuel rods; 79 full length rods, 12 long partial length rods, and one square water rod replacing 9 fuel rods.
19. Contains in total 96 fuel rods; 80 full length rods, 8 long partial length rods, 8 short partial length rods and one water rod replacing 4 or 12 fuel rods.
20. Contains in total 112 fuel rods; 92 full length rods, 8 long partial length rods, 12 short partial length rods, and one square water rod replacing 9 fuel rods.

Table 2.1.4

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	BWR (MPC-89)	PWR (MPC-37)	PWR (MPC-32ML)	PWR (MPC-37P)	PWR (MPC-44)
Reactivity/ Criticality	GE-12/14 10x10 (Array/Class 10x10A)	Westinghouse 17x17 OFA (Array/Class 17x17B)	Siemens 16x16 FOCUS (Array/Class 16x16D)	Combustion Engineering (CE) 15x15 (Array/Class 15x15I)	Westinghouse 14x14 OFA (Array/Class 14x14B)
Shielding	GE-12/14 10x10	Westinghouse 17x17 OFA	Siemens 16x16 FOCUS	Combustion Engineering (CE) 15x15 (Array/Class 15x15I)	Westinghouse 14x14 OFA
Thermal-Hydraulic	GE-12/14 10x10	Westinghouse 17x17 OFA	Siemens 16x16 FOCUS	Combustion Engineering (CE) 15x15 (Array/Class 15x15I)	Westinghouse 14x14 OFA

Table 2.1.6

Soluble Boron Requirements for MPC-37, ~~and~~ MPC-32ML, MPC-37P, and MPC-44 Wet Loading and Unloading Operations

MPC	Array/Class	All Undamaged Fuel Assemblies		One or More Damaged Fuel Assemblies and/or Fuel Debris	
		Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
MPC-37	All 14x14 and 16x16A, B, C, E	1,000	1,600 (Note 3)	1,300	1,800
	All 15x15 and 17x17	1,500	2,000	1,800	2,300
MPC-32ML	16x16D	1,500	2,000	1,600	2,100
MPC-37P	15x15I	1,500	2,000	1,800	2,300
MPC-44	14x14A, B	1,400	1,900	1,500	2,000

Note:
1. For maximum initial enrichments between 4.0 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.0 wt% and 5.0 wt% ^{235}U.
2. If burnup credit is used (as described in Section 2.1.7), these soluble boron requirements do not apply.
3. For 16x16E assembly class, the soluble boron requirement is 1,500 ppmb.

TABLE 2.1.12
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-37P AND MPC-44

Cell Decay Heat Load Limit (kW)	**Polynomial Coefficients, see Paragraph 2.1.6.2**				
	A	**B**	**C**	**D**	**E**
MPC-37P					
≤ 0.79	-7.95196E-18	1.45069E-12	-7.94501E-08	1.81131E-03	-1.09897E+01
0.79 < decay heat ≤ 0.95	-1.25365E-19	2.60073E-13	-2.20748E-08	7.29884E-04	-4.89153E+00
0.95 < decay heat ≤ 1.37	-5.32045E-18	1.07046E-12	-7.62281E-08	2.40535E-03	-2.51659E+01
1.37 < decay heat ≤ 2.64	-1.78154E-21	1.30015E-15	1.74063E-10	1.65882E-05	1.36110E+00
2.64 < decay heat ≤ 3.03	3.36123E-20	2.18480E-15	1.42012E-10	9.23077E-06	6.00000E-01
MPC-44					
≤ 1.0	2.30737E-18	-1.72421E-13	3.59688E-09	9.41169E-05	6.41653E-01

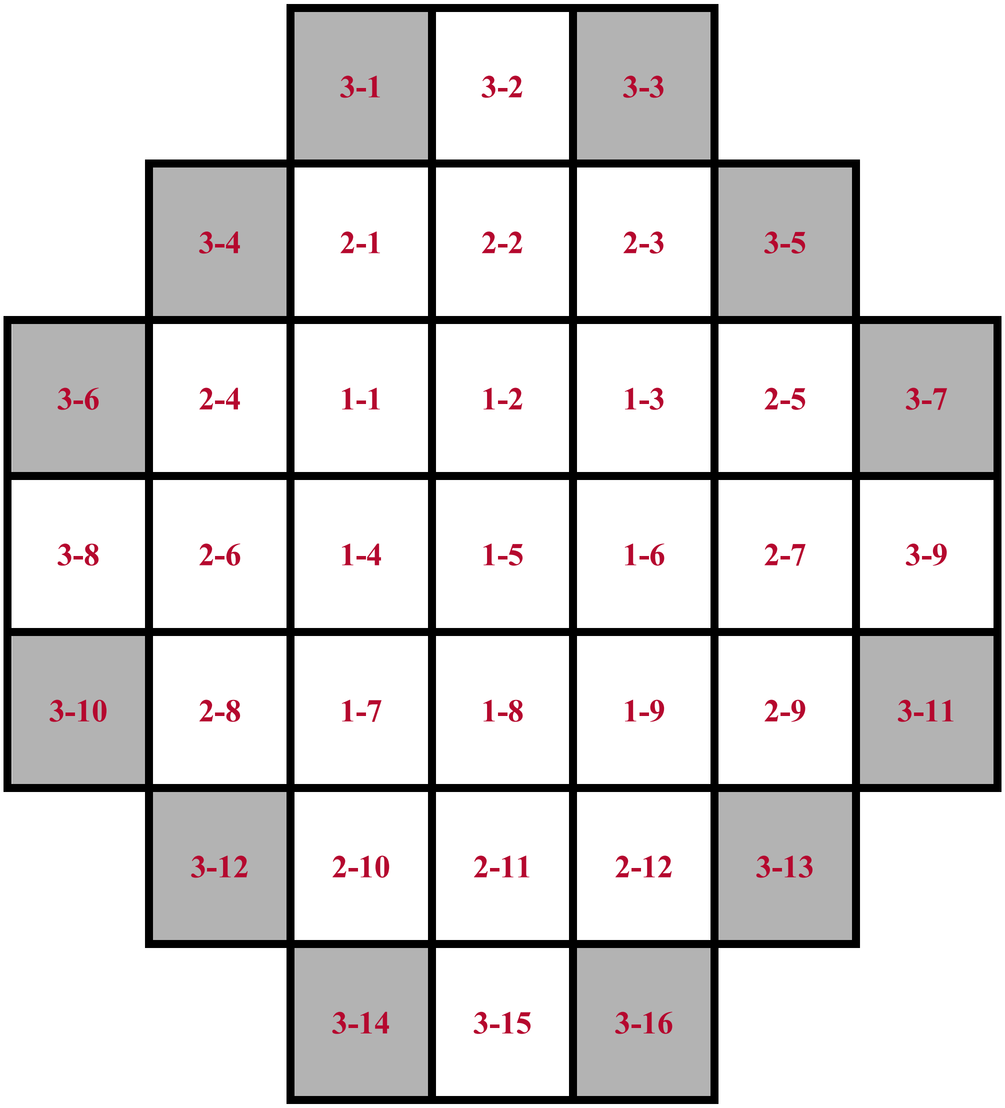


Figure 2.1.1c Location of DFCs for Damaged Fuel or Fuel Debris in the MPC-37P(Shaded Cells)

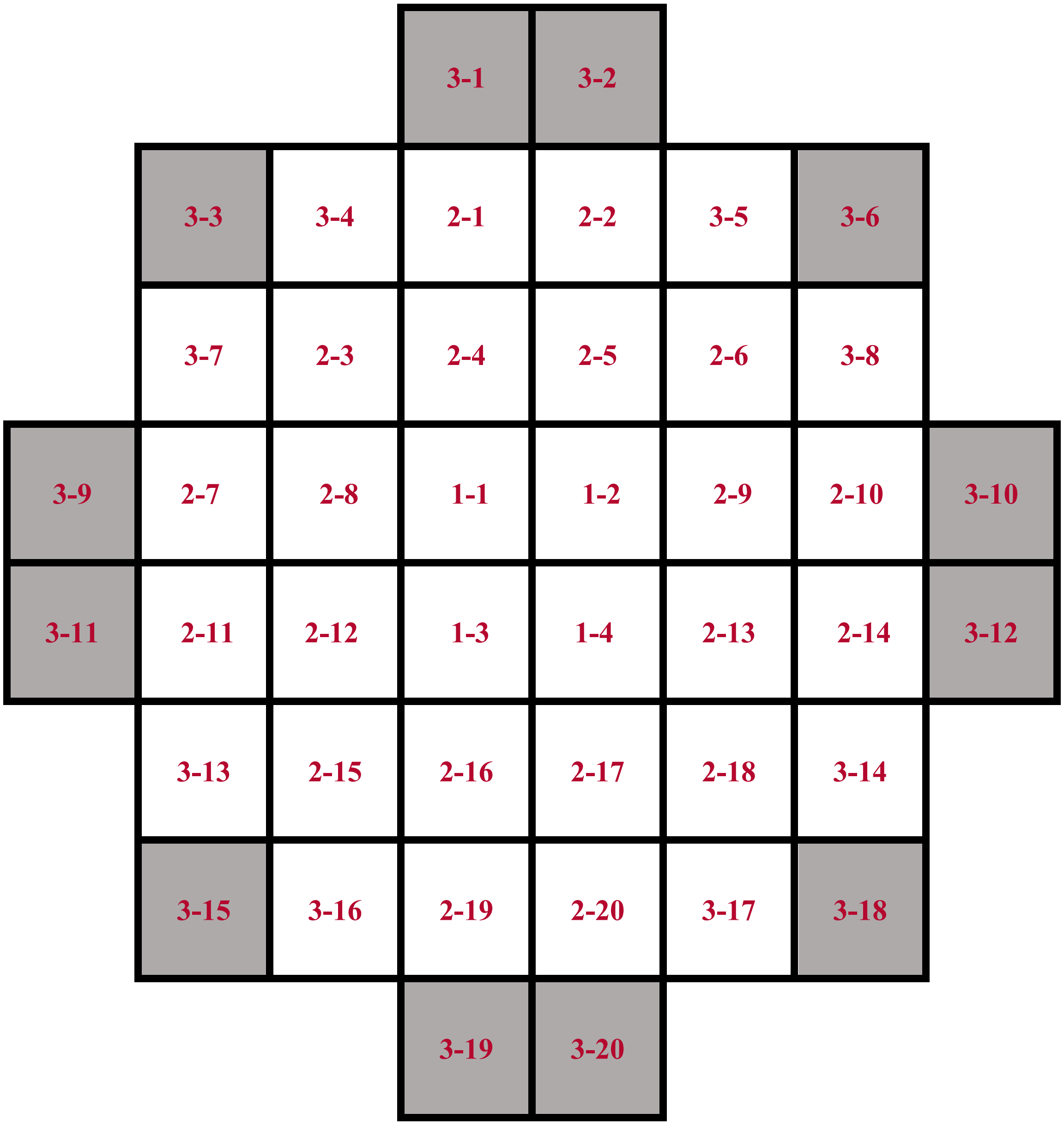


Figure 2.1.1d Location of DFCs for Damaged Fuel or Fuel Debris
in the MPC-44(Shaded Cells)

77°F for Key West, Florida. Therefore, this value is specified in Table 2.2.2 as the bounding soil temperature.

Confirmation of the site-specific annual average ambient temperature and soil temperature is to be performed by the licensee, in accordance with 10CFR72.212. Insolation based on 10CFR71.71 input averaged over 24 hours shall be used as the additional heat input under the normal and off-normal conditions of storage.

The ambient pressure shall be assumed to be 760mm of Hg coincident with the normal condition temperature, whose bounding value is provided in Table 2.2.2. For sites located substantially above sea level (elevation > 1500 feet), it will be necessary to perform a site specific evaluation of the peak cladding temperature using the site specific ambient temperature (maximum average annual temperature based on 40 year meteorological data for the site). ISG 11, Revision 3 [2.0.1] temperature limits will continue to apply.

All of the above requirements are consistent with those in the HI-STORM 100 FSAR.

e. Design Temperatures

The ASME Boiler and Pressure Vessel Code (ASME Code) requires that the value of the vessel design temperature be established with appropriate consideration for the effect of heat generation internal or external to the vessel. The decay heat load from the spent nuclear fuel is the internal heat generation source for the HI-STORM FW System. The ASME Code (Section III, Paragraph NCA-2142) requires the design temperature to be set at or above the maximum through thickness mean metal temperature of the pressure part under normal service (Level A) condition. Consistent with the terminology of NUREG-1536, this temperature is referred to as the "Design Temperature for Normal Conditions". Conservative calculations of the steady-state temperature field in the HI-STORM FW System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STORM FW component temperatures at or below the normal condition design temperatures for the HI-STORM FW System defined in Table 2.2.3.

Maintaining fuel rod cladding integrity is also a design consideration. The fuel rod peak cladding temperature (PCT) limits for the long-term storage and short-term operating conditions shall meet the intent of the guidance in ISG-11, Revision 3 [2.0.1]. For moderate burnup fuel the PCT limit for short-term operations is higher than for high burnup fuel [2.0.2].

f. Snow and Ice

The HI-STORM FW System must be capable of withstanding pressure loads due to snow and ice. Section 7.0 of ANSI/ASCE 7-05 [2.2.3] provides empirical formulas and tables to compute the effective design pressure on the overpack due to the accumulation of snow for the contiguous U.S. and Alaska. Typical calculated values for heated structures such as the HI-STORM FW System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure load (Table 2.2.8) is set to bound the ANSI/ASCE 7-05 recommendation.

g. HI-DRIP Supplemental Cooling System

The HI-DRIP supplemental cooling system is used for the short-term operations when the loaded transfer cask is out of the pool but the MPC contains water. It is not intended for long term operation. No safety analysis is required because there is no credible failure mode for the malfunction of the system since it relies only on the motive pressure provided by the plant's water supply system. If the plant's water supply system were to fail, the time-to-boil of the transfer cask is monitored.

the air flow path is not credible. The HI-TRAC VW Versions V and V2 shall not be left unattended and thus additional analyses are not required.

f. Malfunction of FHD

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.

Initiating events 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 and heat dissipation in the MPC transitions to natural convection cooling.

Although the FHD System is monitored during its operation, stoppage of FHD operations does not require actions to restore forced cooling for adequate heat dissipation. This is because the condition of natural convection cooling evaluated in Section 4.6 shows that the fuel temperatures remain below off-normal limits. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

2.2.3 Environmental Phenomena and Accident Condition Design Criteria

Environmental phenomena and accident condition design criteria are defined in the following subsections.

The minimum acceptance criteria for the evaluation of the accident conditions are that the MPC confinement boundary continues to confine the radioactive material, the MPC fuel basket structure maintains the configuration of the contents, the canister can be recovered from the overpack, and the system continues to provide adequate shielding.

The environmental loads and handling evolutions for a specific site during Part 72 short term operations are influenced by both the architectural layout as well as the geological and meteorological characteristics of the site, and therefore, are apt to be different from the standard handling steps summarized in the chapter on Operations. To perform the site-specific safety analysis of the handling evolutions, the magnitude of the incident load should be informed by its frequency of likely occurrence and the mitigative measures employed. All short-term operations, except the transitional transitory steps (such as upending or down-ending of the cask, placement of the Closure Lid on HI-STORM), shall be subject to safety analysis under the postulated environmental loads. The method of analysis shall follow prior established precedent where available.

A discussion of the effects of each environmental phenomenon and accident condition is provided in Section 12.2. The consequences of each accident or environmental phenomenon are evaluated against the requirements of 10CFR72.106 and 10CFR20. Section 12.2 also provides the corrective action for each event.

i. Prevention of sliding: Assuming the vertical ZPA to be acting to reduce the weight of the cask, horizontal force equilibrium yields:

$$W \cdot a_H \leq \mu \cdot W \cdot (1-a_v)$$

Or $$a_H \leq (1-a_v) \cdot \mu$$

ii. Prevention against "edging" of the cask:

Balancing the moment about the cask's pivot point for edging yields:

$$W \cdot a_H \cdot h \leq W \cdot (1-a_v) \cdot r$$

Or $$a_H \leq (1-a_v) \cdot \frac{r}{h}$$

Where:

- r: radius of the footprint of the cask's base
- h: height of the CG of the cask
- μ: Static friction coefficient between the cask and the ISFSI pad.

The above two inequalities define the limits on a_H and a_v for a site if the earthquake is to be considered of "low intensity." For low intensity earthquake sites, additional analysis to demonstrate integrity of the confinement boundary is not required.

However, if the earthquake's ZPAs do not satisfy either of the above inequalities, then a dynamic analysis using the methodology specified in Chapter 3 shall be performed as a part of the §72.212 safety evaluation.

With respect to short-term operations in dry storage campaigns, a probabilistic risk assessment (PRA) may be employed at a geologically stable and low return frequency site, characterized by a small number of annual loadings, to evaluate the risk associated with a simultaneous earthquake event. However, for defense-in-depth, regardless of the loading campaign's seismic PRA metric, it is necessary to demonstrate a positive margin against overturning of the cask transporter during the translocation of the cask to the ISFSI pad. For this purpose, a static equilibrium analysis of the cask transporter assemblage shall be performed in which it is subject to the site's DBE with its acceleration magnified as suggested in NUREG-0800 (i.e., factor of 1.1). For kinematic stability analysis, the haul path may be simulated as an unyielding surface, and the limiting grade in the haul path incorporated in the solution for conservatism. Safety against potential sliding of the assemblage leading to an impact with a safety significant structure proximate to the haul path shall also be ascertained.

h. <u>100% Fuel Rod Rupture</u>

The HI-STORM FW System must withstand loads due to 100% fuel rod rupture. For conservatism, 100% of the fuel rods are assumed to rupture with 100% of the fill gas and 30% of

Table 2.2.3

TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short-Term Events†† Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits † (°F)
MPC shell	600*	800*	800*
MPC basket	752	932	932
MPC basket shims	752	932	932
MPC lid	600*	800*	800*
MPC closure ring	500*	800*	800*
MPC baseplate	450~~00~~*	800*	800*
HI-TRAC VW inner shell	-	600	700
HI-TRAC VW outer shell	-	500	700
HI-TRAC VW water jacket shell	-	500	700‡‡
HI-TRAC VW bottom lid	-	500	700
HI-TRAC VW top flange	-	500	650
HI-TRAC VW bottom lid seals	-	400	N/A
HI-TRAC VW bottom lid bolts	-	400	800

†† Short term operations include but are not limited to MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

† For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the ISFSI fire event, the local temperature limit of HI-STORM concrete is 1100°F (HI-STORM 100 FSAR Appendix 1.D), and the steel structure is required to remain physically stable (i.e., so there will be no risk of structural instability such as gross buckling, the maximum temperature shall be less than 50% of the component's melting temperature and the specific temperature limits in this table do not apply). Concrete that exceeds 1100°F shall be considered unavailable for shielding of the overpack.

* Temperature limits in Table 1.A.6 shall take precedence if duplex stainless steels are used for the fabrication of confinement boundary components, as described in Appendix 1.A.

‡‡ For fire accidents, the steel structure is required to remain physically stable similar to HI-STORM overpack steel.

Notes: 1. The normal condition temperature limits are used in the design basis structural evaluations for MPC and HI-STORM. The short-term condition temperature limits are used in the design basis structural evaluations for HI-TRAC. All other short-term, off-normal and accident condition structural evaluations are based on bounding temperatures from thermal evaluations presented in Chapter 4.

2. The temperature limits provided for HI-TRAC VW are applicable to Version V and V2 unless otherwise specified.

Table 2.2.3

TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short-Term Events†† Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits † (°F)
HI-TRAC VW bottom flange	-	400	700
HI-TRAC VW radial neutron shield	-	311	N/A
HI-TRAC VW radial lead gamma shield	-	600	600
HI-TRAC VW Version V2 NSC steel	-	400	600
HI-TRAC VW Version V2 NSC Holtite-A	-	300	350
Fuel Cladding	752	752 or 1058 (Short Term Operations)††	1058 (Off-Normal and Accident Conditions)
Overpack concrete	300 (See HI-STORM 100 FSAR Appendix 1.D)	572 (on local temperature of shielding concrete)	572
Overpack Lid Top and Bottom Plate	450	450	572
Overpack Inner Shell	475	700	800
Remainder of overpack steel structure	350	350	700
Damaged Fuel Isolator	752	932	932

TABLE 2.2.14
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

			clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.
MPC Enclosure Vessel	NB-4122	Implies that with the exception of studs, bolts, nuts and heat exchanger tubes, CMTRs must be traceable to a specific piece of material in a component.	MPCs are built in lots. Material traceability on raw materials to a heat number and corresponding CMTR is maintained by Holtec through markings on the raw material. Where material is cut or processed, markings are transferred accordingly to assure traceability. As materials are assembled into the lot of MPCs being manufactured, documentation is maintained to identify the heat numbers of materials being used for that item in the multiple MPCs being manufactured under that lot. A specific item within a specific MPC will have a number of heat numbers identified as possibly being used for the item in that particular MPC of which one or more of those heat numbers (and corresponding CMTRS) will have actually been used. All of the heat numbers identified will comply with the requirements for the particular item.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested. If the redundant port cover design is used, a helium leakage test is not required.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only progressive liquid penetrant (PT) examination is permitted. PT examination will include the root and final weld layers and each approx. 3/8" of weld depth.

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

The HI-STORM FW System is engineered to provide for the safe long-term storage of spent nuclear fuel (SNF). The HI-STORM FW will withstand all normal, off-normal, and postulated accident conditions without release of radioactive material or excessive radiation exposure to workers or members of the public. Special considerations in the design have been made to ensure long-term integrity and confinement of the stored SNF throughout all cask normal and off-normal operating conditions and its retrievability for further processing or ultimate disposal in accordance with 10 CFR 72.122(l) and ISG-2 [2.3.1].

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactivity which the HI-STORM FW System must confine originates from the spent fuel assemblies and, to a lesser extent, any radioactive particles from contaminated water in the fuel pool which may remain inside the MPC. This radioactivity is confined by multiple engineered barriers.

Contamination on the outside of the MPC from the fuel pool water is minimized by preventing contact, removing the contaminated water, and decontamination. An inflatable seal in the annular gap between the MPC and HI-TRAC VW, and the elastomer seal in the HI-TRAC VW bottom lid (see Chapter 9) prevent the fuel pool water from contacting the exterior of the MPC and interior of the HI-TRAC VW while submerged for fuel loading.

The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, MPC shell, MPC lid, closure ring, port cover plates, and associated welds.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, accident conditions, or external natural phenomena. Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity. Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7. MPC field weld examinations, helium leakage testing of the port cover plate welds (if applicable), and pressure testing are performed to ensure the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 10, to verify the integrity of the confinement boundary.

being stored. Basket configurations designed for both PWR and BWR fuel are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell.

- The MPC shell is separated from the basket and its lateral supports (basket shims) by a small, calibrated gap designed to prevent significant thermal stressing associated with the thermal expansion mismatches between the fuel basket, the basket support structure, and the MPC shell. Refer to discussion on DTE earlier in this subsection.

The MPC fuel basket maintains the spent nuclear fuel in a subcritical arrangement. Its safe operation is assured by maintaining the physical configuration of the storage cell cavities intact in the aftermath of a non-mechanistic tipover event. This requirement is satisfied if the MPC fuel basket plates undergo a minimal deflection (see Table 2.2.11). The fuel basket strains are shown in Subsection 3.4.4.1.4 to remain largely elastic with only localized areas of plastic strain. Moreover, from the stimulation results it is demonstrated that the cross section of the storage cell, throughout the active fuel length, remains essentially unchanged. Therefore, there is no impairment in the recoverability or retrievability of the fuel and the subcriticality of the stored fuel is unchallenged.

[PROPRIETARY INFORMATION WITHHELD PER 10CFR2.390]

In normal operating condition, these shims are not subject to any significant loadings. The only condition in which this shim configuration experiences significant loads is the non-mechanistic tipover event when the shim extension plates may be subject to cantilever loads. This loading, which bounds all other events including seismic loads, is considered in the tipover analysis presented in Subsection 3.4.4.1.4b.

Similarly, MPC-44 and MPC-37P are evaluated for non-mechanistic tipover in Subsections 3.4.4.1.4c and 3.4.4.1.4d.

The MPC Confinement Boundary contains no valves or other pressure relief devices. In addition, the analyses presented in Subsections 3.4.3, 3.4.4.1.5, and 3.4.4.1.6 show that the MPC Enclosure Vessel meets the stress intensity criteria of the ASME Code, Section III, Subsection NB for all service conditions. Therefore, the demonstration that the MPC Enclosure Vessel meets Subsection NB stress limits ensures that there will be no discernible release of radioactive materials from the MPC.

(ii) Storage Overpack

The HI-STORM FW storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of un-reinforced (plain) concrete. Plain concrete is also installed in the lid to minimize skyshine. The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a shock absorber for the MPC in the event of a postulated tipover accident. The storage overpack is not a pressure vessel since it contains cooling vents. The structural steel weldment of the HI-STORM FW overpack is designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3 for normal and off-normal loading conditions and Regulatory Guide 3.61 for handling conditions.

- Length: inch

3.1.3.1 HI-STORM FW Overpack

The physical geometry and materials of construction of the HI-STORM FW overpack are provided in Sections 1.1 and 1.2 and the drawings in Section 1.5. The finite element simulation of the overpack consists of two types of models, one for the overpack body and the other for the top lid. Because the loaded overpack is virtually identical in weight and height for the standard, Version XL, domed and Version E with concrete densities up to 200 pcf, the analyses that do not require a detailed simulation of the lid apply to all four configurations. Loading events that require a detailed characterization of the lid's response such as lid lifting and tornado missile impact on the HI-STORM FW lid are analyzed for each lid type separately ([3.4.13] and [3.4.15]). For overpack body with high density concrete (250 pcf), only the weight is different when compared to the standard overpack model.

The models are initially developed using the finite element code ANSYS ([3.4.1] and [3.4.25]), and then, depending on the load case, numerical simulations are performed either in ANSYS or in LS-DYNA [3.1.8]. For example, the handling loads (Load Case 9) and the snow load (Load Case 10) are simulated in ANSYS, and the non-mechanistic tipover event (Load Case 4) is simulated in LS-DYNA. For the non-mechanistic tipover analysis, ~~three~~ distinct finite element models are developed with HI-STORM FW overpack carrying the maximum length MPC-37 (see Figure 3.4.10A) and the maximum length MPC-89 (see Figure 3.4.10B), as well as the MPC-32ML and the MPC-44. The overpack FE model for the MPC-32ML is the same as that for the MPC-37. This conservatively maximizes the weight and the angular velocity of the overpack for the non-mechanistic tipover analysis. The enclosure vessels for the MPC-44 and the MPC-37P are the same as that for the MPC-37.

The key attributes of the HI-STORM FW overpack models (implemented in ANSYS) are:

i. The finite element discretization of the overpack is sufficiently detailed to accurately articulate the primary membrane and bending stresses as well as the secondary stresses at locations of gross structural discontinuity. The finite element layouts of the HI-STORM FW overpack body and the top lid are pictorially illustrated in Figures 3.4.3A/B and 3.4.5A-D, respectively. The overpack model consists of over 70,000 nodes and 50,000 elements, which exceed the number of nodes and elements in the HI-STORM 100 tipover model utilized in [3.1.4]. Table 3.1.11 summarizes the key input data that is used to create the finite element models of the HI-STORM FW overpack body and top lid.

ii. The overpack baseplate, anchor blocks, and the lid studs are modeled with SOLID45 elements. The overpack inner and outer shells, bottom vent shells, and the lifting ribs are modeled with SHELL63 elements. A combination of SOLID45, SHELL63, and SOLSH190 elements is used to model the steel components in the HI-STORM FW standard lid. A combination of SOLID185 and SOLSH190 elements is used to model the steel components in the HI-STORM FW Version XL lid, HI-STORM FW domed lid, and the HI-STORM FW

3.1.3.2 Multi-Purpose Canister (MPC)

The two constituent parts of the MPC, namely (i) the Enclosure Vessel and (ii) the Fuel Basket, are modeled separately. The model for the Enclosure Vessel is focused to quantify its stress and strain field under the various loading conditions. The model for the Fuel Basket is focused on characterizing its strain and displacement behavior during a non-mechanistic tipover event. For the non-mechanistic tipover analysis, ~~three~~ distinct finite element models are created: one for the maximum length MPC-37, one for the maximum length MPC-89, ~~and~~ one for the MPC-32ML and one for the MPC-44. The finite element models for the MPC-37 and MPC-89 enclosure vessels are shown in Figures 3.4.11A and 3.4.11B, respectively. Note that the MPC-32ML enclosure vessel, carrying PWR fuel type, is identical to the MPC-37 except for the length. The finite element models for the fuel baskets, the fuel assemblies and the basket shims, for ~~all~~ three basket types are shown in Figures 3.4.12, 3.4.13 and 3.4.14, respectively. The finite element model for the fuel basket with the fuel assemblies and the basket shims of MPC-44 are described in [3.4.31].

The key attributes of the MPC finite element models (implemented in ANSYS) are:

i. The finite element layout of the Enclosure Vessel is pictorially illustrated in Figure 3.4.1. The finite element discretization of the Enclosure Vessel is sufficiently detailed to accurately articulate the primary membrane and bending stresses as well as the secondary stresses at locations of gross structural discontinuity, particularly at the MPC shell to baseplate juncture. This has been confirmed by comparing the ANSYS stress results with the analytical solution provided in [3.4.16] (specifically Cases 4a and 4b of Table 31) for the discontinuity stress at the junction between a cylindrical shell and a flat circular plate under internal pressure (100 psig). The two solutions agree within 3% indicating that the finite element mesh for the Enclosure Vessel is adequately sized. Table 3.1.14 summarizes the key input data that is used to create the finite element model of the Enclosure Vessel.

ii. The Enclosure Vessel shell, baseplate, and upper and lower lids are meshed using SOLID185 elements. The MPC lid-to-shell weld and the reinforcing fillet weld at the shell-to-baseplate juncture are also explicitly modeled using SOLID185 elements (see Figure 3.4.1).

iii. Consistent with the drawings in Section 1.5, the MPC lid is modeled as two separate plates, which are joined together along their perimeter edge. The upper lid is conservatively modeled as 4.5" thick, which is less than the minimum thickness specified on the licensing drawing (see Section 1.5). "Surface-to-surface" contact is defined over the interior interface between the two lid plates using CONTA173 and TARGE170 contact elements.

iv. The materials used to represent the Enclosure Vessel are assumed to be isotropic and are assigned linear elastic material properties based on the Alloy X material data provided in Section 3.3. The Young's modulus value varies throughout the model based on the applied temperature distribution, which is shown in Figure 3.4.27 and conservatively bounds the temperature distribution for the maximum length MPC as determined by the thermal analyses in Chapter 4 for short-term normal operations.

v. The fuel basket models (Figures 3.4.12A, 3.4.12B and 3.4.12C), which are implemented in

3.2 WEIGHTS AND CENTERS OF GRAVITY

As stated in Chapter 1, while the diameters of the MPC, HI-STORM FW, and HI-TRAC VW are fixed, their height is dependent on the length of the fuel assembly. The MPC cavity height (which determines the external height of the MPC) is set equal to the nominal fuel length (along with control components, if any) plus Δ, where Δ is a small adder provided to account for irradiation and thermal growth of the fuel in the reactor. Table 3.2.1 provides the height of the internal cavities and bottom-to-top external dimension of all system components. Table 3.2.2 provides the parameters that affect the weight of cask components and their range of values assumed in this FSAR.

The cavity heights of the HI-STORM FW overpack and the HI-TRAC VW transfer cask are set greater than the MPC height by fixed amounts to account for differential thermal expansion and manufacturing tolerances. Table 3.2.1 provides the height data on HI-STORM FW, HI-TRAC VW, and the MPC as the adder to the MPC cavity length.

Table 3.2.5 provides the reference weight of the HI-STORM FW overpack for storing MPC-37 and MPC-89 containing reference PWR and BWR fuel, respectively. Conservatively, the HI-STORM FW overpack storing MPC-32ML, MPC-37P and MPC-44 carrying PWR fuel, uses ~~the same~~bounding or maximum PWR fuel ~~reference~~ weights listed in Tables 3.2.5 and 2.1.1 for structural analysis purposes. The weight of the HI-STORM FW overpack body is provided for multiple concrete densities and for two discrete heights for PWR and BWR fuel for both standard, Version XL, and Version E configurations. The weight at any other density and any other height can be obtained by linear interpolation. Similarly, the weight of the HI-STORM FW standard lid, Domed lid, and Version E lid is provided for two discrete values of concrete density. The weight corresponding to any other density can be computed by linear interpolation.

As discussed in Section 1.2, the weight of the HI-TRAC VW transfer cask is maximized for a particular site to take full advantage of the plant's crane capacity within the architectural limitations of the Fuel Building. Accordingly, the thickness of the lead shield and outer diameter of the water jacket can be increased to maximize shielding. The weight of the empty HI-TRAC VW cask in Table 3.2.4 is provided for three lengths corresponding to PWR fuel. Using the data for three lengths, the transfer cask's weight corresponding to any other length can be obtained by linear interpolation (or extrapolation). For MPC-89, the weight data is provided for the minimum and reference fuel lengths, as well as the reference fuel assembly with a DFC and therefore likewise the transfer cask's weight corresponding to any other length can be obtained by linear interpolation (or extrapolation).

The approximate change in the empty weight of HI-TRAC VW (in kilo pounds) of a certain height, h (inch), by virtue of changing the thickness of the lead by an amount, δ (inch), is given by the formula:

$$\Delta W_{lead} = 0.1128(h - 13.5)\,\delta$$

The approximate change in the empty weight of HI-TRAC VW (in kilo pounds) of a certain length,

$$\Delta_r = \frac{\phi D}{100} = \frac{(2.0)(140in)}{100} = 2.8in$$

$$\Delta_v = \frac{\Psi H}{100} = \frac{(\pm 3.0)(207.75in)}{100} = \pm 6.23in \text{ (CG height relative to H/2)}$$

~~The C.G. information provided above shall be used in designing the lifting and handling ancillary for the HI-STORM FW cask components. In addition, t~~The maximum CG height per Table 3.2.7 shall be used for the stability analysis of the HI-STORM FW under DBE conditions unless a more accurate CG height is calculated on a site-specific basis. Using the weight data in the previously mentioned tables, Table 3.2.8 has been constructed to provide the bounding weights for structural analyses so that every load case is analyzed using the most conservative data (to *minimize the computed safety margins*). The weight data in Table 3.2.8 is used in all structural analyses in this chapter.

Table 3.2.8

BOUNDING WEIGHTS FOR STRUCTURAL ANALYSES
(Height from Tables 3.2.1 and 3.2.2)

	Case	Purpose	Assumed Weight (Kilo-pounds)
1.	Loaded HI-STORM FW on the pad containing maximum length/weight fuel and 200 lb/cubic feet concrete	Sizing and analysis of lifting and handling locations and cask stability analysis under overturning loads such as flood and earthquake	425.7 NOTE 2
2.	Loaded HI-STORM FW on the pad with 150 lb concrete, shortest length MPC	Stability analysis under missile strike	285.7
3.	Loaded HI-TRAC VW with maximum length fuel and maximum lead and water shielding	Analysis for NUREG-0612 compliance of lifting and handling locations (TALs and Trunnions)	270.0 NOTE 1
4a.	50% Loaded HI-TRAC VW (tallest cask) ~~with shortest length MPC and minimum lead and water shielding~~	Stability analysis under missile strike	200.2~~183.5~~ NOTE ~~1~~3
4b.	50% Loaded HI-TRAC VW (shortest and lightest cask)		158.5 NOTE 3
5.	Loaded MPC containing maximum length/weight fuel – maximum possible weight scenario	Analysis for NUREG-0612 compliance of lifting and handling locations (TALs)	116.4
6.	Loaded HI-STORM FW on the pad containing reference length/weight fuel and 250 lb/cubic feet concrete – maximum possible weight scenario	Sizing and analysis of lifting and handling locations and cask stability analysis under overturning loads such as flood and earthquake	450.0 NOTE 1

NOTE 1: The listed weight conservatively bounds all HI-TRAC VW versions (maximum or minimum loaded weight, as applicable).

NOTE 2: For users with heavier loaded cask weight, a site-specific evaluation shall be performed for cask lifting and stability.

NOTE 3: For users with lighter loaded cask weight, a site-specific tornado wind and missile evaluation shall be performed for stability; stability analysis assumes that center of mass of stored fuel assemblies lies close to the cask centerline axis.

herein, the method of analysis presented here will provide the means for the site-specific safety evaluation pursuant to 10CFR72.212.

The overturning analysis of the cask under the tornado wind load and large missile impact is performed by solving the 1-DOF equation of motion for the cask angular rotation, which is same methodology used in the HI-STORM 100 FSAR (Docket No. 72-1014). Specifically, the solution of the post-impact dynamics problem is obtained by solving the following equation of motion:

$$I_r \alpha = \left(- W_c \frac{a}{2} \right) + F_{max} \left(\frac{L}{2} \right)$$

where:

I_r	=	cask moment of inertia about the pivot point
α	=	angular acceleration of the cask
W_c	=	lower bound weight of the cask
a/2	=	restoring moment arm = ~~diameter~~ radius of cask at its base (see Figure 3.4.7) at time zero
F_{max}	=	force on the cask due to tornado wind/instantaneous pressure drop
L/2	=	overturning moment arm = half-height of the cask (see Figure 3.4.7) at time zero

In the above equation, α, a, and L are time dependent variables. The impacting missile enters into the above through the post-strike angular velocity of the cask, which is the relevant initial condition for the cask equation of motion. The solution gives the post-impact position of the cask centroid as a function of time, which indicates whether the cask remains stable.

The following assumptions are made in the analysis:

i. The cask is assumed to be a rigid solid cylinder, with uniform mass distribution. This assumption implies that the cask sustains no plastic deformation (i.e. no absorption of energy through plastic deformation of the cask occurs).

ii. The angle of incidence of the missile is assumed to be the worst case of a perfect horizontal impact at the maximum horizontal missile velocity and the angle that maximizes overturning moment arm (see Figure 3.4.7) with the missile traveling at a reduced velocity corresponding to the inclined angle~~such that its overturning effect on the cask is maximized (see Figure 3.4.7)~~.

iii. The analysis considers the maximum height cask and the minimum height cask (with their corresponding weights) per Tables 3.2.1 and 3.2.2. The missile is assumed to strike at the highest point of the cask (see Figure 3.4.7), again maximizing the overturning effect.

iv. The cask is assumed to pivot about a point at the bottom of the base plate opposite the location of missile impact and the application of wind force in order to conservatively

maximize the propensity for overturning (see Figure 3.4.7).

v. Inelastic impact is assumed, with the missile velocity reduced to zero after impact. This assumption conservatively lets the missile impart the maximum amount of angular momentum to the cask, and it is in agreement with missile impact tests conducted by EPRI [3.4.14].

vi. The analysis is performed using the minimum loaded HI-STORM FW weight and 50% loaded HI-TRAC VW weight per Table 3.2.8. A lighter cask will tend to rotate further after the missile strike. The weight of the missile is not included in the total post-impact weight.

vii. Planar motion of the cask is assumed; any loads from out-of-plane wind forces are neglected.

viii. The drag coefficient for a cylinder in turbulent cross flow is used.

ix. The missile and wind loads are assumed to be perfectly aligned in direction.

The results for the post-impact response of the HI-STORM FW overpack and the HI-TRAC VW transfer cask are summarized in Table 3.4.5. The table shows that both casks remain in a vertical upright position (i.e., no overturning) in the aftermath of a large missile impact. The complete details of the tornado wind and large missile impact analyses for the HI-STORM FW overpack and the HI-TRAC VW transfer cask are provided in ~~Appendix 3.A~~[3.4.15].

The results for the post-impact response of the HI-TRAC VW Versions V and V2 transfer casks are summarized in Tables 3.4.5A and 3.4.5B, respectively. The table shows that the cask remains in a vertical upright position (i.e., no overturning) in the aftermath of a large missile impact. The complete details of the tornado wind and large impact analyses for the HI-TRAC VW Versions V and V2 transfer casks are provided in [3.4.15].

Sliding Analysis

A conservative calculation of the extent of sliding of the HI-STORM FW overpack and the HI-TRAC VW cask due to the impact of a large missile (Table 2.2.5) and tornado wind (Table 2.2.4) is obtained using a common formulation as explained below. A more realistic impact simulation using LS-DYNA, with less bounding assumptions, has been used in Subsection 3.4.4.1.4 to qualify the HI-STORM overpack for a non-mechanistic tip over event. While it is not necessary for demonstrating adequate safety margins for this problem, an LS-DYNA analysis could also be used to calculate the sliding potential of the HI-STORM FW and HI-TRAC VW for a large missile impact. In what follows, both HI-STORM FW and HI-TRAC VW are identified by the generic term "cask".

The principal assumptions that render these calculations for sliding conservative are:

i. The weight of the cask used in the analysis is assumed to be the lowest per Table 3.2.8.

through the concrete.

The analyses documented in ~~Appendix 3.B~~[3.4.15] show that the depth of penetration of the small missile is less than the thinnest section of material on the exterior surface of the HI-STORM FW or the HI-TRAC VW. Therefore, the small missile will dent, but not penetrate, the cask. The 1-inch missile can enter the air inlet/outlet vents in the HI-STORM FW overpack, but geometry prevents a direct impact with the MPC.

For the intermediate missile, the analyses documented in ~~Appendix 3.B~~[3.4.15] show that there will be no penetration through the concrete surrounding the inner shell of the HI-STORM FW overpack or penetration of the standard lid. Likewise, the intermediate missile will not penetrate the lead surrounding the HI-TRAC VW inner shell. Therefore, there will be no impairment to the Confinement Boundary due to tornado-borne missile strikes. Furthermore, since the HI-STORM FW and HI-TRAC VW inner shells are not compromised by the missile strike, there will be no permanent deformation of the inner shells and ready retrievability of the MPC will be assured. Similar calculations for an intermediate missile strike on the Version XL, Domed, and Version E lids are performed in [3.4.15].

The penetration results for the small and intermediate missile, for all versions of the HI-STORM FW overpack and lid, are summarized in Table 3.4.6.

The calculations for HI-TRAC VW Versions V and V2 transfer casks are performed in [3.4.15] following the same approach used for HI-TRAC VW transfer cask, and the conclusions are identical to those for impacts from small and intermediate missiles on HI-TRAC VW. The penetration results for HI-TRAC VW Versions V and V2 transfer casks due to impacts from small and intermediate results are summarized in Tables 3.4.6A and 3.4.6B, respectively.

3.4.4.1.4a Load Case 4: Non-Mechanistic Tipover of Standard Basket Design

The non-mechanistic tipover event, as described in Subsection 2.2.3(b), is site-dependent only to the extent that the stiffness of the target (ISFSI pad) affects the severity of the impact impulse. To bound the majority of ISFSI pad sites, the tipover analyses are performed using a stiff target foundation, which is defined in Table 2.2.9. The results presented in the main body of this subsection, including referenced Figures, were obtained using a foundation concrete strength of 6000 psi. Tipover analyses have also been performed using the bounding target foundation concrete strength in Table 2.2.9. The results of these analyses are summarized at the end of this subsection.

The objectives of the analyses are to demonstrate that the plastic deformation in the fuel basket is sufficiently limited to permit the stored SNF to be retrieved by normal means and that there is no significant loss of radiation shielding in the storage system. Furthermore, the maximum lateral deflection of the lateral surface of the fuel basket is within the limit assumed in the criticality analyses (Chapter 6), and therefore, the lateral deflection does not have an adverse effect on criticality safety.

which can be integrated over the limits $\theta_1 = 0$ to $\theta_1 = \theta_{2f}$ (Figure 3.4.8). The final angular velocity θ_1 at the time instant just prior to contact with the ISFSI pad is given by the expression

$$\dot{\theta}_1 (t_B) = \sqrt{\frac{2\,Mgr}{I_A}(1 - \cos\theta_{2f})}$$

where, from Figure 3.4.8,

$$\theta_{2f} = \cos^{-1}\left(\frac{d}{2r}\right)$$

This equation establishes the initial conditions for the final phase of the tip-over analysis; namely, the portion of the motion when the cask is decelerated by the resistive force at the ISFSI pad interface. Using the data germane to HI-STORM FW (Table 3.4.11) and the above equations, the angular velocity of impact is calculated as

$$\dot{\theta}_1 (t_B) = 1.45 \text{ rad/sec}$$

The LS-DYNA analysis to characterize the response of the HI-STORM FW system under the non-mechanistic tipover event is focused on two principal demonstrations, namely:

(i) The lateral deformation of the basket panels in the active fuel region is less than the limiting value in Table 2.2.11.

(ii) The impact between the MPC guide tubes and the MPC does not cause a thru-wall penetration of the MPC shell.

~~Three~~ Four LS-DYNA finite element models are developed to simulate the postulated tipover event of HI-STORM FW storage cask with loaded MPC-37, MPC-44, MPC-89 and MPC-32ML~~, respectively~~. The ~~three~~ LS-DYNA models are constructed according to the dimensions specified in the licensing drawings included in Section 1.5; the tallest configuration for each MPC enclosure type is considered to ensure a bounding tipover analysis. Because of geometric and loading symmetries, a half model of the loaded cask and impact target (i.e., the ISFSI pad) is considered in the analysis. The LS-DYNA models of the HI-STORM FW overpack and the MPC are described in Subsections 3.1.3.1 and 3.1.3.2, respectively. The tipover analysis for MPC-44 is postulated only in the HI-STORM FW Version E overpack.

The ISFSI pad LS-DYNA model, which consists of a 320”×100”×36” concrete pad and the underlying subgrade (800”×275”×470” in size) with non-reflective lateral and bottom surface boundaries, is identical to that used in the HI-STORM 100 tipover analysis documented in the HI-STORM 100 FSAR [3.1.4]. All structural members of the loaded cask are explicitly modeled so that any violation of the acceptance criteria can be found by examining the LS-DYNA simulation results (note: the fuel assembly, which is not expected to fail in a tipover event, is modeled as an elastic rectangular body). This is an improvement compared with the approach taken in the HI-STORM

100 tipover analysis, where the loaded MPC was modeled as a cylinder and therefore the structural integrity of the MPC and fuel basket had to be analyzed separately based on the rigid body deceleration result of the cask. Except for the fuel basket, which is divided into four parts based on the temperature distribution of the basket, each structural member of the cask is modeled as an independent part in the LS-DYNA model. Note that the critical weld connection between the MPC shell and the MPC lid is treated as a separate part and modeled with solid elements. Each of the two LS-DYNA models consists of forty-two parts, which are discretized with sufficiently high mesh density; very fine grids are used in modeling the MPC enclosure vessel, especially in the areas where high stress gradients are expected (e.g., initial impact location with the overpack). To ensure numerical accuracy, full integration thin shell and thick shell elements with 10 though-thickness integration points or multi-layer solid elements are used. The LS-DYNA tipover model consists of over 470,000 nodes and 255,000 elements for HI-STORM FW with loaded MPC-37, and the model for the cask with loaded MPC-89 consists of over 689,000 nodes and 350,000 elements. The tipover model with loaded MPC-32ML consists of 451,310 nodes and 278,646 elements [3.4.28].

The same ISFSI concrete pad material model used for the HI-STORM 100 tipover analysis reported in [3.1.4] is repeated for the HI-STORM FW tipover analysis. Specifically, the concrete pad behavior is characterized using the same LS-DYNA material model (i.e., MAT_PSEUDO_TENSOR or MAT_016) as for the end drop and tipover analyses of the HI-STORM 100 storage cask (the only difference between the HI-STORM FW reference ISFSI concrete pad model and the model of the HI-STORM 100 Set B ISFSI concrete pad is thickness). Moreover, the subgrade is also conservatively modeled as an elastic material as before. Note that this ISFSI pad material modeling approach was originally taken in the USNRC approved storage cask tipover and end drop LS-DYNA analyses [3.4.5] where a good correlation was obtained between the analysis results and the test results.

To assess the potential damage of the cask caused by the tipover accident, an LS-DYNA nonlinear material model with strain rate effect is used to model the responses of all HI-STORM FW cask structural members based on the true stress-strain curves of the corresponding materials. Note that the strain rate effect for the fuel basket material, i.e., Metamic HT, is not considered for conservatism.

Figures 3.4.9A through 3.4.9C depict the three finite-element tipover analysis models developed for the bounding HI-STORM FW cask configurations with loaded MPC-37, MPC-89 and MPC-32ML, respectively. The finite-element analysis model for tipover with MPC-44 fuel basket is described in [3.4.31].

As shown in Figure 3.4.15 and [3.4.31], the fuel basket does not experience significant plastic deformation in the active fuel region to exceed the acceptable limits; plastic deformation is essentially limited locally in cells near the top of the basket beyond the active fuel region for the MPC-37, MPC-44, MPC-89 and MPC-32ML baskets. The fuel basket is considered to be structurally safe since it can continue maintaining appropriate spacing between fuel assemblies after the tipover event. The MPC enclosure vessel experiences minor plastic deformation at the impact locations with the overpack guide tubes; the maximum local plastic strain (10.9%, see Figure 3.4.16) is well below the failure strain of the material and smaller than the plastic strain limit (i.e., at least

iv. The cask closure lid does not dislodge after the tipover event, i.e., the closure lid bolts remain in-tact.

v. The structural analyses of cask closure lids are performed in [3.4.13] using bounding peak deceleration values; therefore, the lids do not suffer any gross loss of shielding.

3.4.4.1.4c Load Case 4: Non-Mechanistic Tipover of MPC-44 Basket Design

The tipover analysis is performed for the MPC-44 basket design using the existing design basis tipover model in LS-DYNA where the MPC-37 standard basket and aluminum shims are replaced with a fully articulated MPC-44 basket. The bounding target foundation properties per Table 2.2.9 are utilized.

The details of the finite element model, input data and results are archived in the calculation package [3.4.30]. The following conclusions demonstrate that all safety criteria are satisfied for the cask system with MPC-44 basket design.

i. The lateral deflection of the most heavily loaded basket panel in the active fuel region complies with the deflection criterion in Table 2.2.11.

ii. The shims remain attached to the basket maintaining its physical integrity.

iii. The plastic strains in the MPC enclosure vessel remain below the allowable material plastic strain limit.

iv. The cask closure lid does not dislodge after the tipover event, i.e., the closure lid bolts remain in-tact.

v. The structural analyses of cask closure lids are performed in [3.4.13] using bounding peak deceleration values; therefore, the lids do not suffer any gross loss of shielding.

3.4.4.1.4d Load Case 4: Non-Mechanistic Tipover of MPC-37P Basket Design

The tipover analysis of HI-STORM FW Version E cask with MPC-37P basket is not explicitly performed because of the following reasons:
a. MPC-37P basket panels are thicker than that of MPC-37 basket per licensing drawings in Section 1.5.
b. MPC-37P basket cell width is smaller than that of MPC-37 basket per licensing drawings in Section 1.5.
c. Weight of MPC-37P fuel assemblies is conservatively bounded by MPC-37 fuel assemblies per Table 2.1.1.
d. Temperature distribution of MPC-37P basket panels is bounded by MPC-37 basket panels per thermal analyses supporting Chapter 4.

The details of the comparative evaluation are documented in [3.4.30]. Therefore, the acceptance criteria defined in Paragraph 2.2.3b are satisfied for HI-STORM FW cask with MPC-37P basket.

3.4.4.1.5 Load Case 5: Design, Short-Term Normal and Off-Normal MPC Internal Pressure

The MPC Enclosure Vessel, which is designed to meet the stress intensity limits of ASME

shown in Figure 3.4.24. The maximum primary stress intensities in the MPC Enclosure Vessel are compared with the applicable stress intensity limits from Subsection NB of the ASME Code [3.4.4]. The allowable stress intensities are taken at 800°F for shell, lid, MPC baseplate, and MPC baseplate-to-shell juncture. These temperatures are obtained from Table 2.2.3 for accident conditions and bound the calculated temperatures under normal operating conditions for the respective MPC components based on the thermal evaluations in Chapter 4. The allowable stress intensities are determined based on normal operating temperatures since the MPC accident internal pressure is dictated by the 100% fuel rod rupture accident, which does not cause any significant rise in MPC temperatures. In fact, the temperatures inside the MPC tend to decrease as a result of the 100% fuel rod rupture accident due to the increase in the density and internal pressure of the circulating gas. The maximum calculated stress intensities in the MPC Enclosure Vessel, and their corresponding allowable limits, are summarized in Table 3.4.8 for Load Case 6.

3.4.4.1.7 Load Case 7: Accident External Pressure

The only affected component for this load case is the MPC Enclosure Vessel. The accident external pressure (Table 2.2.1) is selected sufficiently high to envelop hydraulic-pressure in the case of flood or explosion-induced pressure at all ISFSI Sites.

The main effect of an external pressure on the MPC is to cause compressive stress in the MPC shell. Therefore, the potential of buckling must be investigated. The methodology used for this investigation is from ASME Code Case N-284-2 (Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC (1/07)). This Code Case has been previously used by Holtec in [3.1.4] and accepted by the NRC as a valid method for evaluation of stability in vessels.

The detailed evaluation of the MPC shell under accident external pressure is provided in ~~Appendix 3.C~~Supplement 4 of [3.4.13]. It is concluded that positive safety margins exist so that elastic or plastic instability of the maximum height MPC shell does not occur under the applied pressure.

3.4.4.1.8 Load Case 8: Non-Mechanistic Heat-Up of the HI-TRAC VW Water Jacket

Even though the analyses presented in Chapter 4 indicate that the temperature of water in the water jacket shall not reach boiling and the rupture disks will not open, it is (non-mechanistically) assumed that the hydraulic pressure in the water jacket reaches the relief devices' set point. The objective of this analysis is to demonstrate that the stresses in the water jacket and its welds shall be below the limits set down in an appropriate reference ASME Boiler and Pressure Vessel Code (Section II Class 3) for the Level D service condition. The accident pressure inside the water jacket is given in Table 2.2.1.

The HI-TRAC VW water jacket is analyzed using classical strength-of-materials. Specifically, the unsupported span of the water jacket shell between radial ribs is treated as a curved beam, with clamped ends, under a uniformly distributed radial pressure. The force and moment reactions at the ends of the curved beam for this type of loading are calculated using the formula for Case 5j of Table 18 in [3.4.16]. The primary membrane plus bending stress is then calculated using the formula for

Table 3.4.5

CASK ROTATIONS DUE TO LARGE MISSILE IMPACT

Event	Calculated Value (deg)	Allowable Limit (deg)	Safety Factor
Missile Impact plus Tornado Wind on HI-STORM FW	3.87	29.8	7.70
Missile Impact plus Pressure Drop on HI-STORM FW	4.41	29.8	6.76
Missile Impact plus Tornado Wind on HI-TRAC VW (standard version)	~~14.88~~15.74	23.19	~~1.56~~1.47
Missile Impact plus Pressure Drop on HI-TRAC VW (standard version)	~~12.66~~12.29	23.19	~~1.83~~1.89

Table 3.4.5A

CASK ROTATIONS DUE TO LARGE MISSILE IMPACT – HI-TRAC VW VERSION V

Event	Calculated Value (deg)	Allowable Limit (deg)	Safety Factor
Missile Impact plus Tornado Wind on HI-TRAC VW Version V	15.34~~00~~	23.21	1.51~~5~~
Missile Impact plus Pressure Drop on HI-TRAC VW Version V	~~12.74~~9.60	23.21	~~1.82~~2.42

Table 3.4.5B

CASK ROTATIONS DUE TO LARGE MISSILE IMPACT – HI-TRAC VW VERSION V2

Event	Calculated Value (deg)	Allowable Limit (deg)	Safety Factor
Missile Impact plus Tornado Wind on HI-TRAC VW Version V2	~~16.43~~9.15	21.24	~~1.29~~2.32
Missile Impact plus Pressure Drop on HI-TRAC VW VersionV2	~~13.55~~7.76	21.24	~~1.57~~2.74

Table 3.4.16

CASK SLIDING DISPLACEMENTS DUE TO LARGE MISSILE IMPACT (LOAD CASE 3)

Cask	Calculated Sliding Displacement (ft)	Allowable Sliding Displacement (ft)	Safety Factor
HI-STORM FW	0.454	3.33 (cask to cask) 6.2 (cask to edge of ISFSI pad)	7.33 13.6
HI-TRAC VW	1.319~~133~~	None Established	-
HI-TRAC VW Version V	1.193	None Established	-
HI-TRAC VW Version V2	0.958	None Established	-

v. The load combinations for normal, off-normal, accident, and natural phenomena events have been compiled and applied on the MPC Enclosure Vessel (Confinement Boundary). The results, summarized in Section 3.4, show that the factor of safety (with respect to the appropriate limits) is greater than one in all cases. Design Basis natural phenomena events such as tornado-borne missiles (large, intermediate, or small) have also been analyzed to evaluate their potential for reaching and breaching the Confinement Boundary. Analyses presented in Section 3.4 ~~and supplemented by Appendices 3.A and 3.B~~ show that the integrity of the Confinement Boundary is preserved under all design basis projectile impact scenarios.

- The information on structural design included in this FSAR complies with the requirements of 10CFR72.120 and 10CFR72.122.

- The structural design features in the HI-STORM FW system are in compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

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[3.4.26] ANSYS 17.1, SAS IP, Inc., 2016.

[3.4.27] Carette and Malhotra, Performance of Dolostone and Limestone Concretes at Sustained High Temperatures, Temperature Effects on Concrete, ASTM STP 858, 1985, p. 38-67.

[3.4.28] Holtec Proprietary Report HI-2166998, "Analysis of the Non-Mechanistic Tipover Event of the Loaded HI-STORM FW Storage Cask Loaded with MPC-32ML and MPC-31C", Latest Revision.

[3.4.29] Holtec Proprietary Report HI-2188483, "Structural Analysis of Lifting of HI-STORM FW Through Jacking Arrangement", Latest Revision.

[3.4.30] Holtec Proprietary Report HI-2200503, "Analysis of the Non-Mechanistic Tipover Event of the Loaded HI-STORM FW Version E Storage Cask", Latest Revision.

[3.4.31] Holtec Proprietary Report HI-2210313, "Analysis of the Non-Mechanistic Tipover Event of the Loaded HI-STORM FW Version UVH Storage Cask", Latest Revision.

[3.6.1] Visual Nastran 2004, MSC Software, 2004.

CHAPTER 4* THERMAL EVALUATION

4.0 OVERVIEW

The HI-STORM FW system is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. The design envisages an array of HI-STORM FW systems laid out in a rectilinear pattern stored on a concrete ISFSI pad in an open environment. In this chapter, compliance of HI-STORM FW system's thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI using 3-D thermal simulation models is established. The analyses consider passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. 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 FW system are quantified. Safe thermal performance during on-site loading, unloading and transfer operations, collectively referred to as "short-term operations" utilizing the HI-TRAC VW transfer cask is also evaluated.

The HI-STORM FW thermal evaluation follows the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4]. 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 must meet the temperature limit under normal, off-normal and accident conditions appropriate to its burnup level and condition of storage or handling set forth in Table 4.3.1.
2. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions set forth in Table 2.2.1.
3. The temperatures of the cask materials shall remain below their allowable limits set forth in Table 2.2.3 under all scenarios.

As demonstrated in this chapter, the HI-STORM FW 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 of storage, handling and on-site transfer operations. All thermal analyses to evaluate normal conditions of storage in a HI-STORM FW storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC VW 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 SAR chapter is in full

* 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. All terms-of-art used in this chapter are consistent with the terminology of the Glossary. Finally, all evaluations and results presented in this Chapter are supported by calculation packages cited herein (References [4.1.9], [4.4.4], [4.1.10] and [4.1.12]).

† Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Glossary).

compliance with ISG-11 and with NUREG-1536 guidelines, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.3.

As explained in Section 1.2, the storage of SNF in the fuel baskets, with the exception of MPC-44, in the HI-STORM FW system is configured for a three-region storage system under regionalized storage and uniform storage. The storage of SNF in MPC-44 is configured for a uniform single region storage system. Figures 1.2.1a, 1.2.1b, 1.2.1c, 1.2.1d, and 1.2.2 provide the information on the location of the regions and Tables 1.2.3a, 1.2.3b, 1.2.3c, 1.2.3e, and 1.2.4a provide the permissible specific heat load (heat load per fuel assembly) in each region for the PWR and BWR MPCs, respectively. The Specific Heat Load (SHL) values under regionalized storage are defined for two patterns that in one case maximizes ALARA (Table 1.2.3a, Pattern A and Table 1.2.4a) and in the other case maximizes heat dissipation (Table 1.2.3a, Pattern B). The ALARA maximized fuel loading is guided by the following considerations:

- Region 1: Located in the core region of the basket is permitted to store fuel with medium specific heat load.
- Region 2: This is the intermediate region flanked by the core region (Region I) from the inside and the peripheral region (Region III) on the outside. This region has the maximum SHL in the basket.
- Region 3: Located in the peripheral region of the basket, this region has the smallest SHL. Because a low SHL means a low radiation dose emitted by the fuel, the low heat emitting fuel around the periphery of the basket serves to block the radiation from the Region II fuel, thus reducing the total quantity of radiation emanating from the MPC in the lateral direction.

Thus, the 3-region arrangement defined above serves to minimize radiation dose from the MPC and peak cladding temperatures mitigated by avoiding placement of hot fuel in the basket core.

To address the needs of cask users having high heat load fuel inventories, fuel loading Pattern B is defined in Table 1.2.3a to maximize heat dissipation by locating hotter fuel in the cold peripheral Region 3 and in this manner minimize cladding temperatures. This has the salutary effect of minimizing core temperature gradients in the radial direction and thermal stresses in the fuel and fuel basket.

As an alternative to the loading patterns discussed above, fuel storage in the MPC-37 and MPC-89 is permitted to use the heat load charts shown in Figures 1.2.3a, 1.2.4a, 1.2.5a (MPC-37) and Figures 1.2.6a, 1.2.7a (MPC-89) or Figures 1.2.3b, 1.2.3c, 1.2.4b, 1.2.4c, 1.2.5b, 1.2.5c (MPC-37) and Figures 1.2.6b, 1.2.7b (MPC-89) for damaged fuel and fuel debris in certain locations. MPC-44 is permitted to use only uniform heat load pattern presented in Table 1.2.3e, while MPC-37P is permitted to use the heat load patterns presented in Table 1.2.3c, along with Patterns A and B used for MPC-37 defined in Table 1.2.3a.

The salutary consequences of all regionalized loading arrangements become evident from the computed peak cladding temperatures in this chapter, which show margin to the ISG-11 limit discussed earlier.

Low thermal stresses are also ensured by an MPC design that permits unrestrained axial growth of the basket and results in none to minimal basket-to-shell interference radially. The provision of adequate basket-to-shell gaps results in insignificant thermal stresses due to radial restraint on basket periphery thermal growth as discussed in Paragraph 3.1.1(i) and Subparagraph 3.1.2.2(a).

The most important contributor to minimizing thermal stresses and maximizing heat transmission within the fuel basket is its material of construction (Metamic-HT) which has approximately ten times the thermal conductivity of the stainless steel material used in the stainless steel baskets in the HI-STORM 100 System [4.1.8]. The Metamic-HT plates in the HI-STORM FW MPCs are also considerably thicker than their counterparts in the stainless baskets, resulting in an additional enhancement in conduction heat transfer.

The MPCs uniform & regionalized fuel storage scenarios are defined in Figures 1.2.1a, 1.2.1b, 1.2.1c, 1.2.1d, and 1.2.2 in Chapter 1 and design maximum decay heat loads for storage of zircaloy clad fuel are listed in Tables 1.2.3a, 1.2.3b, 1.2.3c, 1.2.3d, 1.2.3e, 1.2.4a, and in Figures 1.2.3a thru 1.2.3c, 1.2.4a thru 1.2.4c, 1.2.5a thru 1.2.5c, 1.2.6a thru 1.2.6b, 1.2.7a thru 1.2.7b, and 1.2.9. The axial heat distribution in each fuel assembly is conservatively assumed to be non-uniformly distributed with peaking in the active fuel mid-height region (see axial burnup profiles in Figures 2.1.3 and 2.1.4). Table 4.1.1 summarizes the principal operating parameters of the HI-STORM FW system.

The fuel cladding temperature limits that the HI-STORM FW system is required to meet are discussed in Section 4.3 and given in Table 2.2.3. Additionally, when the MPCs are deployed for storing High Burnup Fuel (HBF) further restrictions during certain fuel loading operations (vacuum drying) are set forth herein 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.

The HI-STORM FW system (i.e., HI-STORM FW overpack, HI-TRAC VW transfer cask and MPC) is evaluated under normal storage (HI-STORM FW overpack), during off-normal and accident events and during short-term operations in a HI-TRAC VW. Results of HI-STORM FW thermal analysis during normal (long-term) storage are obtained and reported in Section 4.4. Results of HI-TRAC VW short-term operations (fuel loading, on-site transfer and vacuum drying) are reported in Section 4.5. Results of off-normal and accident events are reported in Section 4.6.

Table 4.1.1

HI-STORM FW OPERATING CONDITION PARAMETERS

Condition	Value
MPC Decay Heat, max.	Tables 1.2.3a thru 1.2.3c~~, 1.2.3b~~, and 1.2.4a Figures 1.2.3a thru 1.2.3c, Figures 1.2.4a thru 1.2.4c, Figures 1.2.5a thru 1.2.5c, Figures 1.2.6a thru 1.2.6b and Figures 1.2.7a thru 1.2.7b, Figure 1.2.9a and 1.2.9b
MPC Operating Pressure	Note 1
Normal Ambient Temperature	Table 2.2.2
Helium Backfill Pressure	Table 4.4.8
Note 1: The MPC operating pressure used in the thermal analysis is based on the minimum helium backfill pressure specified in Table 4.4.8 and MPC cavity average temperature.	

4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

The HI-STORM FW Storage System (i.e., HI-STORM FW overpack and MPC) and HI-TRAC VW transfer cask thermal evaluation is performed in accordance with the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4]. To ensure a high level of confidence in the thermal evaluation, 3-dimensional models of the MPC, HI-STORM FW overpack and HI-TRAC VW transfer cask are constructed to evaluate fuel integrity under normal (long-term storage), off-normal and accident conditions and in the HI-TRAC VW transfer cask under short-term operation and hypothetical accidents. The principal features of the thermal models are described in this section for HI-STORM FW and Section 4.5 for HI-TRAC VW. Thermal analyses results for the long-term storage scenarios are obtained and reported in this section. The evaluation addresses the design basis thermal loadings defined in Section 1.2.3. Based on these evaluations the limiting thermal loading condition is defined in Subsection 4.4.4 and adopted for evaluation of on-site transfer in the HI-TRAC (Section 4.5) and off-normal and accident events defined in Section 4.6.

4.4.1 Overview of the Thermal Model

As illustrated in the drawings in Section 1.5, 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 Metamic-HT plates that are slotted and arrayed in an orthogonal configuration to form an integral basket structure. The Metamic-HT neutron absorber plates contain 10% (min.) Boron Carbide in an aluminum matrix reinforced with nanoparticles of alumina to provide criticality control, while maximizing heat conduction capabilities (see Chapter 1, Section 1.2.1.4.1).

Thermal analysis of the HI-STORM FW System is performed for all heat load scenarios defined in Chapter 1 for regionalized storage in MPC-37 and MPC-89 (Figures 1.2.1a and 1.2.2) and uniform storage in MPC-32ML (Figure 1.2.1b).– Thermal evaluations are also performed for MPC-44 and MPC-37P for the heat load patterns shown in Table 1.2.3e and Table 1.2.3c respectively. It is to be noted that MPC-37P is qualified for use only with HI-STORM FW Version E system, and therefore, the thermal model for MPC-37P is built with the Version E overpack. In addition, MPC-37P basket has higher heat rejection capabilities than MPC-37 basket due to increased basket thickness. Therefore, temperatures and pressures in MPC-37P for heat load patterns A and B (Table 1.2.3a), will be bounded by those for MPC-37. Each fuel assembly is *assumed to be generating heat at the maximum permissible rate (Tables 1.2.3a, 1.2.3b, 1.2.3c, 1.2.3e, 1.2.4a, Figures 1.2.3a thru 1.2.3c, 1.2.4a thru 1.2.4c, 1.2.5a thru 1.2.5c, 1.2.6a thru 1.2.6b, 1.2.7a thru 1.2.7b, 1.2.9a and 1.2.9b)*. While the assumption of limiting heat generation in each storage cell imputes a certain symmetry to the cask thermal problem, it grossly overstates the total heat duty of the system in most cases because it is unlikely that any basket would be loaded with fuel emitting heat at their limiting values in *each* storage cell. Thus, the thermal model for the HI-STORM FW system is inherently conservative for real life applications. Other noteworthy features of the thermal analyses are:

i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange increases rapidly as the fourth power of absolute temperature.

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 lowest values reached at the periphery of the basket.

As noted in Chapter 1 and in Section 3.2, the height of the PWR MPC cavity can vary within a rather large range to accommodate spent nuclear fuel of different lengths in the MPC-37 and MPC-89[1]. The heat load limits in Tables 1.2.3a and 1.2.3b (PWR MPC) and Table 1.2.4 (BWR MPC) for regionalized storage are, however, fixed regardless of the fuel (and hence MPC cavity) length. Because it is not a'priori obvious whether the shortest or the longest fuel case will govern, thermal analyses are performed for the minimum[2], reference and maximum height MPCs. Table 2.1.1 allows two different fuel assembly lengths under "minimum" category for PWR fuel. Unless specified in this chapter, the term "minimum" or "short" is used for all short fuel assembly arrays except 15x15I short fuel defined in Chapter 2.

Various HI-STORM FW overpack designs, namely, the standard, XL and E versions, are developed for long-term storage, as discussed in Chapter 1. Similarly, as described in Chapter 1, the HI-STORM lid is available in three versions. The HI-STORM FW overpack is equipped with thru-wall penetrations at the bottom of the overpack. The exit air passageway is located in the body of the standard lid. The "Version XL" (extra-large) lid and "Domed" lid are variants of the standard lid design with the exit air passageway located at the bottom of the lid near the cask body interface to permit efficient, natural circulation of air to cool the MPC and the contained SNF. The Version E overpack is a variant of the standard overpack with an alternate design of the thru-wall penetrations at the bottom of the overpack, and this version uses a designated lid (Version E lid) which is similar to the Version XL lid. The cask overpack and lid design details are shown in the licensing drawing package (Section 1.5). Similarly, the Version XL lid and the domed lid can be placed only with the Version XL HI-STORM FW overpack – this ensures the axial gap between the MPC and HI-STORM lid is cognizant with Table 3.2.1. Thermal evaluations of HI-STORM FW with standard lid, Version XL lid, Version E and domed lid designs are performed in this chapter and discussed below. Unless stated, the thermal evaluations in this chapter are based on standard HI-STORM FW lid and overpack design.

The evaluations of HI-STORM FW with standard lid, Version XL lid, Version E lid, and domed lid show that the variation in the lid design has small impact on the fuel cladding temperature.

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1 MPC-32ML, MPC-37P and MPC-44 are~~is~~ ~~a~~ fixed fuel length canisters for storing a specific fuel defined in Table 2.1.1b, 2.1.1c, and 2.1.1d respectively.

2 Both allowable PWR fuel assembly lengths under "minimum" category as shown in Table 2.1.1 are evaluated in this chapter.

For HI-STORM FW with standard lid, the optional heat shield on the overpack inner shell and underneath the overpack lid helps lower the temperature of the concrete. On the other hand, the domed lid is equipped with a heat shield plate to protect the concrete, which is similar to the design in the standard lid.

4.4.1.1 Description of the 3-D Thermal Model

i. Overview

The HI-STORM FW System may be equipped with ~~three~~ five MPC designs, MPC-37, MPC-37P, MPC-44, MPC-32ML, and MPC-89 engineered to store 37, 44, 32 PWR, or ~~and~~ 89 BWR fuel assemblies ~~respectively~~. 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]. Because the fuel basket is made of a single isotropic material (Metamic-HT), the 3-D thermal model requires no idealizations of the fuel basket structure. However, since it is 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 cell cross-section is discussed under item (ii) below. 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). The methodology to represent the spent fuel storage space as a homogeneous region with equivalent conductivities is identical to that used in the HI-STORM 100 Docket No. 72-1014 [4.1.8].

ii. Details of the 3-D Model

The HI-STORM FW fuel basket is modeled in the same manner as the model described in the HI-STAR 180 SAR (NRC Docket No. 71-9325) [4.1.11]. Modeling details are provided in the following:

Fuel Basket 3D Model
The MPC-37, MPC-37P, MPC-44, MPC-32ML, and MPC-89 fuel baskets are essentially ~~an~~ arrays of square cells within ~~an~~ irregularly shaped basket outlines. The fuel basket is confined inside a cylindrical cavity of the MPC shell. In the standard MPC configuration, thick Aluminum basket shims are installed in the fuel basket-to-shell spaces to facilitate heat dissipation. To ensure an adequate representation of the fuel basket a geometrically accurate 3D model of the array of square cells and Metamic-HT plates is constructed using the FLUENT pre-processor.

Other than the representation of fuel assemblies inside the storage cell spaces as porous region with effective thermal-hydraulic properties as described in the next paragraph, the 3D model includes an explicit articulation of other canister parts. The basket shims are explicitly modeled in the peripheral spaces. The fuel basket is surrounded by the MPC shell and outfitted with a solid welded lid above and a baseplate below. All of these physical details are explicitly articulated in a quarter-symmetric 3D thermal model of the HI-STORM FW as shown in Figures 4.4.2a, 4.4.2b and 4.4.3.

Fuel Region Effective Planar Conductivity

In the HI-STORM FW thermal modeling, the cross section bounded by the inside of a PWR storage cell and the channeled area of a BWR storage cell is replaced with an "equivalent" square section characterized by an effective thermal conductivity in the planar and axial directions. Figure 4.4.1 pictorially illustrates this concept. The two conductivities are unequal because while in the planar direction heat dissipation is interrupted by inter-rod gaps; in the axial direction heat is dissipated through a continuous medium (fuel cladding). The equivalent planar conductivity of the storage cell space is obtained using a 2D conduction-radiation model of the bounding PWR and BWR fuel storage scenarios defined in the table below. The fuel geometry, consisting of an array of fuel rods with helium gaps between them residing in a storage cell, is constructed using QA validated codes (FLUENT[1] [4.1.2]) and lowerbound conductivities under the assumed condition of stagnant helium (no-helium-flow-condition) are obtained. In the axial direction, an area-weighted average of the cladding and helium conductivities is computed. Axial heat conduction in the fuel pellets is conservatively ignored.

The effective fuel conductivity is computed under ~~four~~ several bounding fuel storage configurations for PWR fueled MPC-37, MPC-32ML for specific PWR fuel and one bounding scenario for BWR fueled MPC-89. The fuel storage configurations are defined below:

Storage Scenario	MPC	Fuel
PWR: 15x15I Short Fuel	Minimum Height MPC-37, MPC-37P for 15x15I fuel assembly array	15x15I in Table 2.1.2
PWR: Short Fuel	Minimum Height MPC-37 for all fuel assembly arrays except 15x15I	14x14 Ft. Calhoun
PWR: Standard Fuel	Reference Height MPC-37	W-17x17
PWR: XL Fuel	Maximum Height MPC-37	AP1000
PWR: 16x16D	MPC-32ML	16x16D
PWR: 14x14	MPC-44	14x14B
BWR	MPC-89	GE-10x10

The fuel region effective conductivity is defined as the calculated equivalent conductivity of the fuel storage cell due to the combined effect of conduction and radiation heat transfer in the manner of the approach used in the HI-STORM 100 system (Docket No. 72-1014). Because

1 NRC has accepted FLUENT code for evaluation of fuel conductivities in the HI-STAR 180D licensing (Docket No. 71-9367).

radiation is proportional to the fourth power of absolute temperature, the effective conductivity is a strong function of temperature. FLUENT computer code has been used to characterize fuel resistance at several representative storage cell temperatures and the effective thermal conductivity as a function of temperature obtained for all storage configurations defined above and tabulated in Table 4.4.1. The effective planar thermal conductivity of fuel for the CBS basket designs are documented in [4.1.10] and [4.1.12].

Heat Rejection from External Surfaces

The exposed surfaces of the HI-STORM FW dissipate heat by radiation and external natural convection heat transfer. Radiation is modeled using classical equations for radiation heat transfer (Rohsenow & Hartnett [4.2.2]). Jakob and Hawkins [4.2.9] recommend the following correlations for natural convection heat transfer to air from heated vertical and horizontal surfaces:

Turbulent range:

$$h = 0.19\,(\Delta T)^{1/3}\ (\text{Vertical}, GrPr > 10^9)$$

$$h = 0.18\,(\Delta T)^{1/3}\ (\text{Horizontal Cylinder}, GrPr > 10^9)$$

(in conventional U.S. units)

Laminar range:

$$h = 0.29\,\left(\frac{\Delta T}{L}\right)^{1/4}\ (\text{Vertical}, GrPr < 10^9)$$

$$h = 0.27\,\left(\frac{\Delta T}{D}\right)^{1/4}\ (\text{Horizontal Cylinder}, GrPr < 10^9)$$

(in conventional U.S. Units)

where ΔT is the temperature differential between the cask's exterior surface and ambient air and GrPr is the product of Grashof and Prandtl numbers. During storage conditions, the cask cylinder and top surfaces are cooled by natural convection. The corresponding length scales L for these surfaces are the cask diameter and length, respectively. As described in Section 4.2, Gr×Pr can be expressed as $L^3\Delta TZ$, where Z (from Table 4.2.7) is at least 2.6×10^5 at a conservatively high surface temperature of 340°F. Thus the turbulent condition is always satisfied assuming a lowerbound L (8 ft) and a small ΔT (~10°F).

Determination of Solar Heat Input

The intensity of solar radiation incident on exposed surfaces depends on a number of time varying parameters. The solar heat flux strongly depends upon the time of the day as well as on latitude and day of the year. Also, the presence of clouds and other atmospheric conditions (dust, haze, etc.) can significantly attenuate solar intensity levels. In the interest of conservatism, the

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 models for both the BWR and PWR storage geometries are constructed for the Design Basis fuel defined in Table 2.1.4. The model contains comprehensive details of the fuel which includes grid straps, BWR water rods and PWR guide and instrument tubes (assumed to be plugged for conservatism).

c) The effective conductivities of the MPC storage spaces are computed for bounding fuel storage configurations defined in Paragraph 4.4.1.1(ii). The in-plane thermal conductivities are obtained using FLUENT [4.4.2] computer models of an array of fuel rods enclosed by a square box. Radiation heat transfer from solid surfaces (cladding and box walls) is enabled in these models. Using these models the effective conduction-radiation conductivities are obtained and reported in Table 4.4.1. The effective planar thermal conductivity of fuel for the CBS basket designs are documented in [4.1.10] and [4.1.12]. For heat transfer in the axial direction an area weighted mean of cladding and helium conductivities are computed (see Table 4.4.1). 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. In the interest of conservatism, thermal analysis of normal storage condition in HI-STORM FW and normal onsite transfer condition in HI-TRAC VW (Section 4.5) are performed with a 10% reduced effective thermal conductivity of fuel region.

d) The internals of the MPC, including the basket cross-section, aluminum shims, bottom flow holes, top plenum, and circumferentially irregular downcomer formed by the annulus gap in the aluminum shims (standard design) or between the basket and inner diameter of the enclosure vessel (CBS design) are modeled explicitly. For simplicity, the flow holes are modeled as rectangular openings with an understated flow area.

e) The inlet and outlet vents in the HI-STORM FW 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 FW/MPC annulus is simulated by the k-ω turbulence model with the transitional option enabled. The adequacy of this turbulence model is confirmed in the Holtec benchmarking report [4.1.6]. The annulus grid size is selected to ensure a converged solution. (See Section 4.4.1.6).

g) A limited number of fuel assemblies defined in Table 1.2.1 classified as damaged fuel are permitted to be stored in the MPC inside Damaged Fuel Containers (DFCs) or Damaged Fuel Isolators (DFIs). A DFC or DFI can be stored in the outer

or Damaged Fuel Isolators (DFIs). A DFC or DFI can be stored in the outer peripheral locations of MPC-37, ~~-~~MPC-32ML, MPC-37P,~~-~~ MPC-44, and MPC-89 as shown in Figures 2.1.1a~~,~~, 2.1.1b, 2.1.1c, 2.1.1d, and 2.1.2, respectively. Additionally, a DFC or DFI can be stored in outer peripheral locations or in certain interior locations as shown in Figures 1.2.3a thru 1.2.3c, 1.2.4a thru 1.2.4c, 1.2.5a thru 1.2.5c, 1.2.6a thru 1.2.6b and 1.2.7a thru 1.2.7b. DFC or DFI emplaced fuel assemblies have a higher resistance to helium flow because of the debris screens. DFC/DFI fuel storage under peripherally permitted scenarios does not affect temperature of hot fuel stored in the core of the basket because DFC/DFI storage is located away from hot fuel. For these scenarios reason the thermal modeling of the fuel basket under the assumption of all storage spaces populated with intact fuel is justified. Interior permitted DFC/DFI storage scenarios are addressed under item "m" below.

h) As shown in HI-STORM FW drawings in Section 1.5 the HI-STORM FW overpack is equipped with an optional heat shield to protect the inner shell and concrete from radiation heating by the emplaced MPC. The inner and outer shells and concrete are explicitly modeled. All the licensing basis thermal analyses explicitly include the heat shields. A sensitivity study is performed as described in paragraph 4.4.1.9 to evaluate the absence of heat shield on the overpack inner shell and overpack lid.

i) To maximize lateral resistance to heat dissipation in the fuel basket, 0.8 mm full length inter- panel gaps are conservatively assumed to exist at all intersections. This approach is identical to that used in the thermal analysis of the HI-STAR 180 Package in Docket 71-9325. The shims installed in the MPC peripheral spaces (See MPC-37, MPC-37P, MPC-44, MPC-32ML and MPC-89 drawings in Section 1.5) are explicitly modeled. For conservatism bounding as-built gaps (3 mm basket-to-shims and 3 mm shims-to-shell) are assumed to exist and incorporated in the thermal models.

j) The thermal models incorporate all modes of heat transfer (conduction, convection and radiation) in a conservative manner.

k) The Discrete Ordinates (DO) model, previously utilized in the HI-STAR 180 docket (Docket 71-9325), is deployed to compute radiation heat transfer.

l) Laminar flow conditions are applied in the MPC internal spaces to obtain a lowerbound rate of heat dissipation.

m) A limited number of fuel assemblies classified as damaged or fuel debris placed in Damaged Fuel Containers (DFCs) or damaged fuel in Damaged Fuel Isolators (DFIs) are permitted to be stored in certain interior locations of MPC-37 and MPC-89 under heat load charts defined in Figures 1.2.3b thru 1.2.3c, 1.2.4b thru 1.2.4c, 1.2.5b thru 1.2.5c, 1.2.6b and 1.2.7b. These scenarios are evaluated herein.

The 3-D model described above is illustrated in the cross-section for the MPC-89, MPC-32ML, ~~and~~ MPC-37, MPC-37P, and MPC-44 in Figures 4.4.2a, 4.4.2b, ~~and~~ 4.4.3, 4.4.8, and 4.4.9 respectively. A closeup of the fuel cell spaces which explicitly include the channel-to-cell gap in the 3-D model applicable to BWR fueled basket (MPC-89) is shown in Figure 4.4.4. The principal 3-D modeling conservatisms are listed below:

1) The storage cell spaces are loaded with high flow resistance design basis fuel assemblies (See Table 2.1.4).
2) Each storage cell is generating heat at its limiting value under the regionalized storage scenarios defined in Chapter 2, Section 2.1.
3) Axial dissipation of heat by conduction in the fuel pellets is neglected.
4) Dissipation of heat from the fuel rods by radiation in the axial direction 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) Reasonably bounding solar absorbtivity of HI-STORM FW overpack external surfaces is applied to the thermal models.
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) Conservatively specified fuel basket emissivity in Table 1.2.8b adopted in the thermal analysis.
12) Lowerbound stainless steel emissivity obtained from cited references (See Table 4.2.1) are applied to MPC shell.
13) The k-ω model used for simulating the HI-STORM FW annulus flow yields uniformly conservative results [4.1.6].
14) Fuel assembly length is conservatively modeled equal to the height of the fuel basket.

The effect of crud resistance on fuel cladding surfaces has been evaluated and found to be negligible [4.1.8]. 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) [4.1.8]. Accordingly this effect is neglected in the thermal evaluations.

iv. Principal Attributes of MPC-32ML 3D Thermal Model

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factors. In the case of BWR fueled MPC-89 the most resistive GE-10x10 fuel assembly in the channeled configuration is explicitly modeled in the MPC-89 fuel storage spaces as shown in Figure 4.4.4. The channeled space occupied by the GE-10x10 fuel assembly is modeled as a porous region with effective flow resistance properties computed by deploying an independent 3D FLUENT model of the array of fuel rods and grid spacers.

In the PWR fuel resistance modeling case physical reasoning suggests that the flow resistance of a fuel assembly placed in the larger MPC-37 storage cell will be less than that computed using the (smaller) counterpart cells cavities in the MPC-32. However to provide numerical substantiation FLUENT calculations are performed for the case of W-17x17 fuel placed inside the MPC-32 cell opening of 8.79” and the enlarged MPC-37 cell opening of 8.94”. The FLUENT results for the cell pressure drops under the baseline (MPC-32) and enlarged cell opening (MPC-37) scenarios are shown plotted in Figure 4-4-7. The plot shows that, as expected, the larger cell cross section case (MPC-37) yields a smaller pressure loss. Therefore, the MPC-37 flow resistance is bounded by the MPC-32 flow resistance used in the FLUENT simulations in the SAR. This evaluation is significant because the MPC-37 basket is determined as the limiting MPC and therefore the licensing basis HI-STORM FW temperatures by use of higher-than-actual resistance are overstated.

However, as mentioned in Sub-section 4.4.1.2, a flow resistance of $1x10^6$ m^{-2} through PWR fuel assemblies is used in the thermal analysis.

4.4.1.5 Screening Calculations to Ascertain Limiting Storage Scenario

To define the thermally most limiting HI-STORM FW storage scenario the following cases are evaluated under the limiting heat load patterns defined in Tables 1.2.3a[1] and 1.2.4:

(i) MPC-89 under regionalized fuel storage Table 1.2.4a
(ii) Minimum height MPC-37 under regionalized fuel storage Table 1.2.3a
(iii) Reference height MPC-37 under regionalized fuel storage Table 1.2.3a
(iv) Maximum height MPC-37 under regionalized fuel storage Table 1.2.3a
(v) MPC-89 under heat load Figures 1.2.6a, 1.2.6b
(vi) MPC-37 under heat load Figures 1.2.3a/b/c, 1.2.4a/b/c and 1.2.5a/b/c
(vii) MPC-89 under heat load Figures 1.2.7a, 1.2.7b
(viii) MPC-37P under heat load Table 1.2.3c
(ix) MPC-44 under heat load Table 1.2.3e

To evaluate the above scenarios, 3D FLUENT screening models of the HI-STORM FW cask are constructed, Peak Cladding Temperatures (PCT) computed and tabulated in Table 4.4.2. The results of the calculations yield the following:

(a) Fuel storage in MPC-37 produces a higher peak cladding temperature than that in

1 Pattern A defined in Table 1.2.3a is the limiting fuel storage pattern (See Subsection 4.4.4.1).

MPC-89, MPC-37P, and MPC-44.

(b) Fuel storage in the minimum height MPC-37 is limiting (produces the highest peak cladding temperature).

To bound the HI-STORM FW storage temperatures the limiting scenario ascertained above is adopted for evaluation of all normal, off-normal and accident conditions.

4.4.1.6 Grid Sensitivity Studies

To achieve grid independent CFD results, a grid sensitivity study is performed on the HI-STORM FW thermal model. The grid refinement is performed in the entire domain i.e. for both fluid and solid regions in both axial and radial directions. Non-uniform meshes with grid cells clustered near the wall regions are generated to resolve the boundary flow near the walls.

A number of grids are generated to study the effect of mesh refinement on the fuel and component temperatures. All sensitivity analyses were carried out for the case of MPC-37 with minimum fuel length under the bounding heat load pattern A. Following table gives a brief summary of the different sets of grids evaluated and PCT results.

Mesh No	Total Mesh Size	PCT (°C)	Permissible Limit (°C)	Clad Temperature Margin (°C)
1 (Licensing Basis Mesh)	1,536,882	373	400	27
2	3,354,908	372	400	28
3	7,315,556	372	400	28
Note: Because the flow field in the annulus between MPC shell and overpack inner shell is in the transitional turbulent regime, the value of y^+ at the wall-adjacent cell is maintained on the order of 1 to ensure the adequate level of mesh refinement is reached to resolve the viscosity affected region near the wall.				

As can be seen from the above table, the PCT is essentially the same for all the meshes. The solutions from the different grids used are in the asymptotic range. Therefore, it can be concluded that the Mesh 1 is reasonably converged. To provide further assurance of convergence, the sensitivity results are evaluated in accordance with the ASME V&V 20-2009 [4.4.3]. Towards this end, the Grid Convergence Index (GCI), which is a measure of the solution uncertainty, is computed to be 0.181% for these meshes. The apparent order of the method calculated as 2.1, is similar to the order of the method.

Based on the above results, Run No 1 grid layout is adopted for the thermal analysis of the HI-STORM FW. The thermal evaluations of MPC-32ML are performed using a mesh with similar mesh density as the licensing basis converged mesh for MPC-37.

4.4.1.7 Evaluation of 15x15I Short Fuel Assembly

The results show that the presence of surrounding casks has essentially no effect on the fuel cladding temperatures (the difference in the results is within the range of numerical round-off). These results are in line with prior thermal evaluations of the effect of surrounding casks in the NRC approved HI-STORM 100 System in Docket 72-1014.

4.4.3 Test Model

The HI-STORM FW 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 FW thermal evaluations, the FLUENT code has been benchmarked using data from tests conducted with casks loaded with irradiated SNF ([4.1.3],[4.1.7]). The benchmark work is archived in QA validated Holtec reports ([4.1.5],[4.1.6]).These evaluations show that the FLUENT solutions are conservative in all cases. In view of these considerations, additional experimental verification of the thermal design is not necessary. FLUENT has also been used in all Holtec International Part 71 and Part 72 dockets since 1996.

4.4.4 Maximum and Minimum Temperatures

4.4.4.1 Maximum Temperatures

(i) Evaluation of Standard HI-STORM FW Lid

The 3-D model from the previous subsection is used to determine temperature distributions under long-term normal storage conditions for both BWR canisters (MPC-89) and PWR canisters (MPC-37, ~~and~~ MPC-32ML, and MPC-44). Tables 4.4.2, 4.4.3 and 4.4.5 provide key thermal and pressure results from the FLUENT simulations, respectively. Tables 4.4.12 and 4.4.13 respectively provide the temperature and pressure results from the FLUENT simulation of the 15x15I short fuel assembly height based on the methodology discussed in Sub-Section 4.4.1.7. The peak fuel cladding result in these tables is actually overstated by the fact that the 3-D FLUENT cask model incorporates the effective conductivity of the fuel assembly sub-model. Therefore the FLUENT models report the peak temperature *in the fuel storage cells.* 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 modest margin.

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

- The fuel cladding temperatures are below the regulatory limit (ISG-11 [4.1.4]) under all uniform and regionalized storage scenarios defined in Chapter 1 (Figures 1.2.1a, 1.2.1b and 1.2.2) and thermal loading scenarios defined in Tables 1.2.3a, 1.2.3b, 1.2.4a, Figures 1.2.3a/b/c, 1.2.4a/b/c, 1.2.5a/b/c, 1.2.6a/b, and 1.2.7a/b.

Table 4.4.4 presents a summary of the MPC free volumes determined for the lowerbound height MPC-89, lowerbound height MPC-37, MPC-37P, MPC-44, and MPC-32ML fuel storage scenarios. 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. A concomitant effect of rod ruptures is the increased pressure and molecular weight of the cavity gases with enhanced rate of heat dissipation by internal helium convection and lower cavity temperatures. As these effects are substantial under large rod ruptures the 100% rod rupture accident is evaluated with due credit for increased heat dissipation under increased pressure and molecular weight of the cavity gases. 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 4.4.5. The results of the calculations support the following conclusions:

(i) The maximum computed gas pressures reported in Table 4.4.5 under all design basis thermal loadings defined in Section 4.4 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Table 2.2.1.

(ii) The MPC gas pressure obtained under loading Pattern A is essentially same as in Pattern B. Accordingly Pattern A loading condition for pressure boundary evaluation of MPC in the HI-TRAC and under off-normal and accident conditions is retained.

Evaluation of Non-Fuel Hardware

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the PWR basket 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 FW 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. The MPC cavity free space is computed based on conservatively computed volume displacement by fuel 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 temperatures 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. They 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 BPRA containing fuel assembly. For a bounding evaluation the maximum permissible number of BPRA containing fuel assemblies (see discussion at the beginning of this Section) are assumed to be loaded. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.5 for the bounding MPC-37 (minimum MPC height and heat load Patterns A and B), MPC-37P, MPC-44, MPC-32ML, and MPC-89 (design heat load) storage scenarios. The CBS design results in larger cavity free volume than the standard basket design. The computed cavity pressure for the CBS design is therefore bounded by that presented in Table 4.4.5.

4.4.6 Engineered Clearances to Eliminate Thermal Interferences

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) differential thermal expansion and (ii) non-uniform temperature distribution. In this subsection, differential thermal expansion calculations are performed to demonstrate that the HI-STORM FW System is engineered with gaps for the fuel basket and the MPC to expand thermally in axial and radial directions without resulting in significant thermal stresses. 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

The FLUENT thermal model provides the 3-D temperature field in the HI-STORM FW system from which the changes in the above gaps are directly computed. Table 4.4.6 provides the initial minimum gaps and their corresponding value during long-term storage conditions. Significant margins against radial and axial expansion of MPC and axial expansion of basket are available in the design. The potential small basket-to-shell interference in the radial direction due to DTE in the basket, shims, and shell is acceptable as discussed in Paragraph 3.1.1(i) and Subparagraph 3.1.2.2(a).

Table 4.4.1

EFFECTIVE FUEL PROPERTIES UNDER BOUNDING FUEL STORAGE CONFIGURATIONS[Note 1]

	Conductivity (Btu/hr-ft-°F)			
	PWR: Short Fuel		**PWR: Standard Fuel**	
Temperature (°F)	Planar	Axial	Planar	Axial
200	0.265	0.802	0.26	0.755
450	0.441	0.891	0.419	0.84
700	0.7	1.002	0.649	0.945
	0.833@800 °F	1.16@1000 °F	0.767@800 °F	1.094@1000 °F
	PWR: XL Fuel		**BWR Fuel**	
	Planar	Axial	Planar	Axial
200	0.269	0.794	0.321	1.077
450	0.426	0.882	0.491	1.189
700	0.647	0.993	0.727	1.332
	0.795@800 °F	1.148@1000 °F	0.847@800 °F	1.539@1000 °F
PWR: 15x15I Short Fuel			**PWR 14x14D Fuel**	
Temperature (°F)	Planar	Axial	Planar ~~Axial~~	Axial
200	0.249	0.760	0.252	0.752
450	0.407	0.845	0.421	0.837
700	0.631	0.952	0.664	0.942
	0.742@800 °F	1.101@1000 °F	1.048@1000°F	1.090@1000°F
PWR: 15x15I in MPC-37P basket				
Temperature (°F)		Temperature (°F)		Temperature (°F)
200		200		200
450		450		450
700		700		700
800		800		800

Thermal Inertia Properties		
	Density (lb/ft^3)	Heat Capacity (Btu/lb-°F)Note 2
PWR: 14x14D Fuel	188.7	0.056
PWR: 15x15I Short Fuel	196.4	0.056
PWR: 15x15I in MPC-37P	203.6	0.056
PWR: Short Fuel	165.1	0.056
PWR: Standard Fuel	175.4	0.056
PWR: XL Fuel	186.6	0.056
BWR Fuel	255.5	0.056

Note 1: Bounding fuel storage configurations defined in 4.4.1.1(ii).
Note 2: The lowerbound heat capacity of principal fuel assembly construction materials tabulated in Table 4.2.5 (UO_2 heat capacity) is conservatively adopted.
Note 3: The fuel properties tabulated herein are used in screening calculations to define the limiting scenario for fuel storage (See Table 4.4.2).
(continued to next page)

Table 4.4.2

RESULTS OF SCREENING CALCULATIONS UNDER NORMAL STORAGE CONDITIONS

Storage Scenario	Peak Cladding Temperature, °C (°F)
MPC-37 - regionalized storage Table 1.2.3a Minimum Height[1] Reference Height Maximum Height	 **353 (667)** 342 (648) 316 (601)
MPC-37 (Note 4) - heat load Figure 1.2.3a - heat load Figure 1.2.4a - heat load Figure 1.2.5a - heat load Figure 1.2.3b[Notes 5,6,7]	 371 (700) 368 (694) 367 (693) 364 (687)
MPC-89 - regionalized storage Table 1.2.4a	 333 (631)
MPC-89 (Note 4) - heat load Figure 1.2.6a - heat load Figure 1.2.6b[Note 5, 7] - heat load Figure 1.2.7a - heat load Figure 1.2.7b[Note 5, 7]	 366 (691) 360 (680) 365 (689) 358 (676)
MPC-37P heat load pattern in Table 1.2.3c	373 (703)
MPC-44, Uniform Heat Load	368 (694)

Notes:

(1) The highest temperature highlighted above is reached under the case of minimum height MPC-37 designed to store the short height Ft. Calhoun 14x14 fuel. This scenario is adopted in Chapter 4 for the licensing basis evaluation of fuel storage in the HI-STORM FW system. See Note 4

(2) All the screening calculations for MPC-37 and MPC-89 were performed using a reference coarse mesh [4.1.9] and flow resistance based on the calculations in Holtec report [4.4.2].

(3) Not Used.

(4) Screening evaluation used the same mesh as licensing basis mesh adopted in Section 4.4.1.6. The computed temperatures are bounded by the licensing basis minimum height temperatures tabulated in Table 4.4.3.

(5) PCT of intact fuel assemblies in the loading patterns with fuel debris in the DFCs is bounded by that with damaged fuel in the DFCs as justified next. It is conservatively assumed that the damaged fuel assemblies inside DFCs/DFIs have the same axial heat distribution as the intact fuel assemblies to maximize the PCT of intact fuel assemblies. Fuel debris consistent with it's physical condition is modeled as packed towards bottom of the DFCs. This yields less impact on the PCT of intact fuel assemblies.

(6) The computed temperature under short length Damaged Fuel Storage is bounded by undamaged fuel temperatures computed above in heat load Figure 1.2.3a. This reasonably supports the conclusion that Damaged Fuel Storage under standard and long fuel storage in Figures 1.2.4b/c, 1.2.5b/c is bounded by undamaged fuel heat load scenarios evaluated in Figure 1.2.4a and 1.2.5a above.

1 Bounding scenario adopted in this Chapter for all thermal evaluations.

(7) Peak temperatures including damaged fuel in DFC/DFI tabulated herein.

Table 4.4.3

MAXIMUM TEMPERATURES IN HI-STORM FW UNDER LONG-TERM NORMAL STORAGE[1]

Component	MPC-37 Temperature, °C (°F) Pattern A / Pattern B	MPC-32ML Temperature, °C (°F)	MPC-44 Temperature, °C (°F)	MPC-37P Temperature, °C (°F)
Fuel Cladding	373 (703) / 368 (694)	349 (660)	368 (694)	373 (703)
MPC Basket	358 (676) / 354 (669)	334 (633)	356 (673)	318 (604)
Basket Periphery	290 (554) / 292 (558)	271 (520)	296 (565)	288 (550)
Aluminum Basket Shims	267 (513) / 267 (513)	256 (493)	283 (541)	271 (520)
MPC Shell	240 (464) / 242 (468)	217 (423)	266 (511)	238 (460)
MPC Lid[Note 1]	235 (455) / 232 (450)	220 (428)	237 (459)	243 (469)
Overpack Inner Shell	126 (259) / 127 (261)	108 (226)	122 (252)	156 (313)
Overpack Outer Shell	65 (149) / 65 (149)	63 (145)	62 (143)	77 (171)
Overpack Body Concrete[Note 1]	89 (192) / 90 (194)	80 (176)	89 (192)	99 (210)
Overpack Lid Concrete[Note 1]	111 (232) / 112 (234)	102 (216)	105 (221)	123 (253)
Area Averaged Air outlet[2]	103 (217) / 103 (217)	99 (210)	107 (225)	91 (196)
Note 1: Maximum section average temperature is reported.				

1 The temperatures reported in this table (all for short fuel scenarios of MPC-37) are below the design temperatures specified in Table 2.2.3, Chapter 2. These temperatures bound MPC-89 temperatures.

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

Table 4.4.4

MINIMUM MPC FREE VOLUMES

Item	**Lowerbound Height MPC-37 (ft^3)**		**MPC-89 (ft3)**
Net Free Volume*	211.89		203.58
	MPC-32ML (ft^3)	**MPC-44 (ft^3)**	**MPC-37P (ft^3)**
Net Free Volume*	278.7	236.1	208.2

**Net free volumes are obtained by subtracting basket, fuel, aluminum shims, spacers, basket supports and DFCs metal volume from the MPC cavity volume.*

Table 4.4.5

SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE*

Condition	**MPC-37 *** (psig) Pattern A/Pattern B**	**MPC-89 *** (psig)**
Initial maximum backfill** (at 70°F)	45.5/46.0	47.5
Normal: intact rods 1% rods rupture	 96.6/97.9 97.7/99.0	 98.4 99.0
Off-Normal (10% rods rupture)	107.5/108.9	104.0
Accident (100% rods rupture)	191.5/194.4	156.9

* 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 4.4.8).

*** Tabulated pressures bound storage under heat load Figures 1.2.3a/b/c, 1.2.4a/b/c, 1.2.5a/b/c, 1.2.6a/b, 1.2.7a/b.

(continued next page)

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Table 4.4.5 (continued)

SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE*

Condition	MPC-32ML (psig)	MPC-44 (psig)	MPC-37P (psig)
Initial maximum backfill** (at 70°F)	45.5	44.0	47.0
Normal: intact rods 1% rods rupture	 91.8 91.1	 91.8 92.9	 96.4 97.2
Off-Normal (10% rods rupture)	98.7	102.3	104.5
Accident (100% rods rupture)	167.5	183.6	177.5

* 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 4.4.8).

Table 4.4.6

SUMMARY OF HI-STORM FW DIFFERENTIAL THERMAL EXPANSIONS

Gap Description	Cold Gap U (in)	Differential Expansion δ_i (in)	Is Free Expansion Criterion Satisfied (i.e., $U > \delta_i$)
Fuel Basket-to-MPC Radial Gap	0.0625	0.082	No*
Fuel Basket-to-MPC Minimum Axial Gap	1.5	0.421	Yes
MPC-to-Overpack Radial Gap	2.625	0.161	Yes
MPC-to-Overpack Minimum Axial Gap	3.5	0.381	Yes
*While the free expansion criterion is not satisfied, the resultant impact to the design is insignificant. See section 4.4.6 for additional details.			

Table 4.4.7
THEORETICAL LIMITS* OF MPC HELIUM BACKFILL PRESSURE**

MPC	Minimum Backfill Pressure (psig)	Maximum Backfill Pressure (psig)
MPC-37 Pattern A	41.0	47.3
MPC-37 Pattern B	40.8	47.1
MPC-37 Figures 1.2.3a Figures 1.2.4a Figure 1.2.5a	 43.9 43.6 44.1	 50.6 50.3 50.8
MPC-89 Table 1.2.4a Figure 1.2.6a Figure 1.2.7a	 41.9 41.7 41.8	 48.4 48.2 48.3

MPC-32ML	39.7	50.6

~~44.047.0~~* The helium backfill pressures are set forth in the Technical Specifications with a margin (see Table 4.4.8).

** The pressures tabulated herein are at 70ºF reference gas temperature.

Table 4.4.8
MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS

MPC	Item	Specification
MPC-37 Pattern A	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	45.5 psig @ 70°F Reference Temperature
MPC-37 Pattern B	Minimum Pressure	41.0 psig @ 70°F Reference Temperature
	Maximum Pressure	46.0 psig @ 70°F Reference Temperature
MPC-89 Table 1.2.4a	Minimum Pressure	42.5 psig @ 70°F Reference Temperature
	Maximum Pressure	47.5 psig @ 70°F Reference Temperature
MPC-32ML	Minimum Pressure	41.5 psig @ 70°F Reference Temperature
	Maximum Pressure	45.5 psig @ 70°F Reference Temperature
MPC-37 Figures 1.2.3a/b/c	Minimum Pressure	45.5 psig @ 70°F Reference Temperature
	Maximum Pressure	49.0 psig @ 70°F Reference Temperature
MPC-37 Figures 1.2.4a/b/c	Minimum Pressure	44.0 psig @ 70°F Reference Temperature
	Maximum Pressure	47.5 psig @ 70°F Reference Temperature
MPC-37 Figures 1.2.5a/b/c	Minimum Pressure	44.5 psig @ 70°F Reference Temperature
	Maximum Pressure	48.0 psig @ 70°F Reference Temperature
MPC-89 Figures 1.2.6a/b	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	47.0 psig @ 70°F Reference Temperature
MPC-89 Figures 1.2.7a/b	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	47.0 psig @ 70°F Reference Temperature
MPC-44	Minimum Pressure	41.0 psig @ 70°F Reference Temperature
	Maximum Pressure	44.0 psig @ 70°F Reference Temperature
MPC-37P Figures 1.2.9a/b	Minimum Pressure	44.0 psig @ 70°F Reference Temperature
	Maximum Pressure	47.0 psig @ 70°F Reference Temperature

Table 4.4.17

DESIGN OPERATING ABSOLUTE PRESSURES[Note 1]

MPC-37 Loading Pattern A Loading Pattern B	 7.1 atm 7 atm
MPC-32ML	6.5 atm
MPC-89 Table 1.2.4a	7 atm
MPC-37 load Figure 1.2.3a	7.0 atm
MPC-37 heat load Figure 1.2.4a	6.9 atm
MPC-37 heat load Figure 1.2.5a	6.8 atm
MPC-89 heat load Figure 1.2.6a	6.8 atm
MPC-89 heat load Figure 1.2.7a	6.8 atm
MPC-44 heat load Table 1.2.3e	6.8 atm
MPC-37P heat load Table 1.2.3c	7.0 atm

Note 1: Table 4.4.8 helium backfill specifications ensure MPC operating pressures meet or exceed design values tabulated herein.

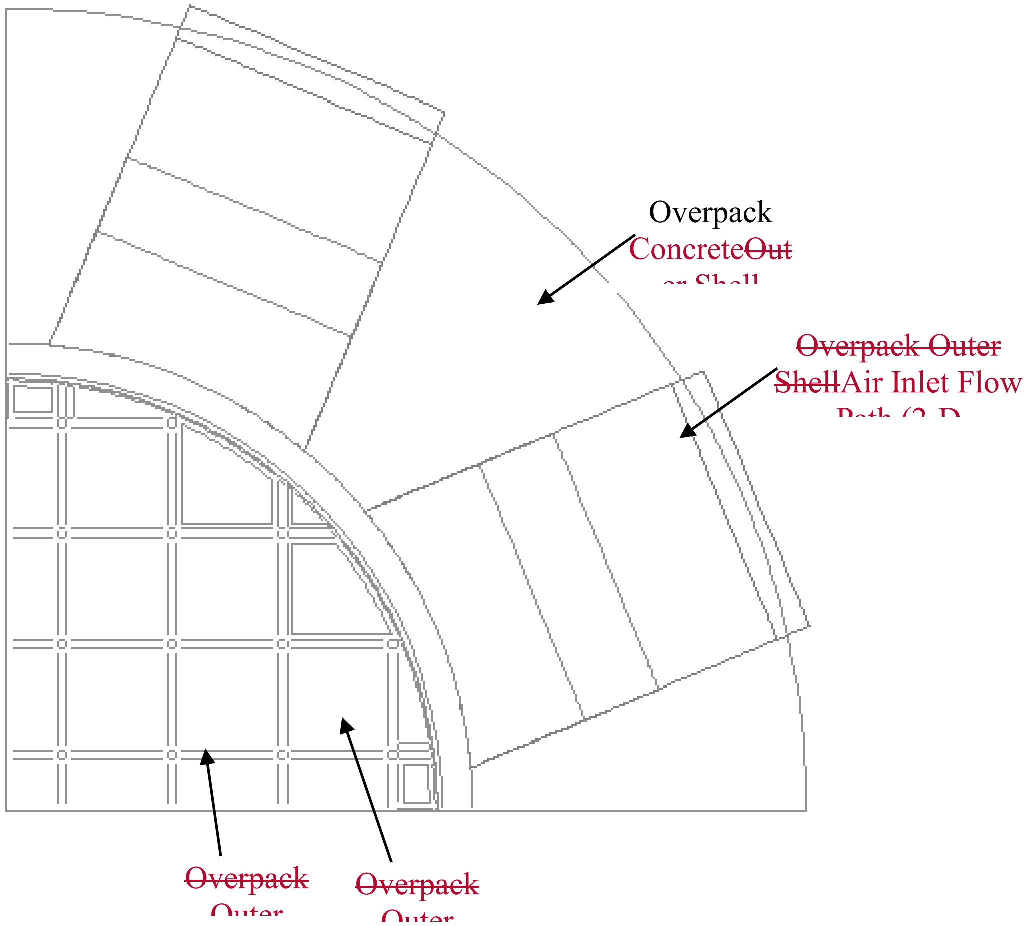

Figure 4.4.8 Planar View of HI-STORM FW Version E MPC-37P Quarter Symmetric 3-D Model

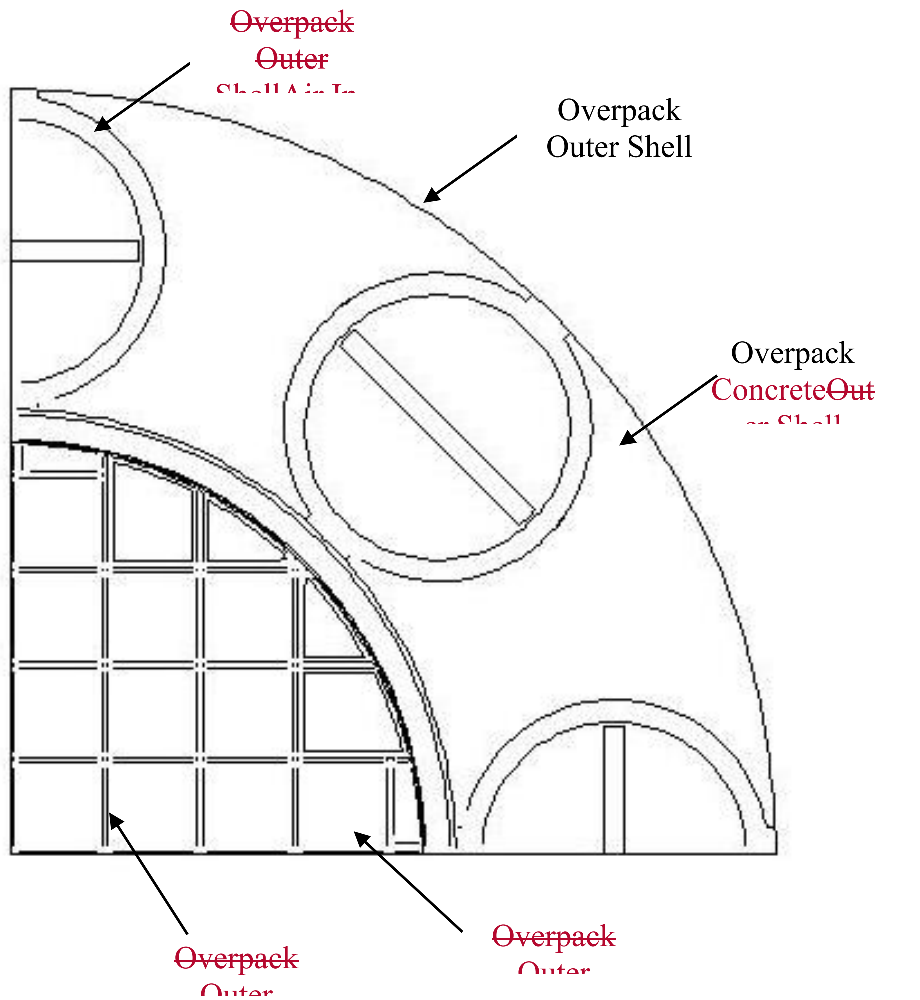

Figure 4.4.9 Planar View of HI-STORM FW MPC-44 Quarter Symmetric 3-D Model

The solutions from these grids are in the asymptotic range. The finest mesh (Mesh 3) has about 4.6 times the total mesh size of the licensing basis mesh (Mesh 1). Even with such a large mesh refinement, the PCT is essentially same for all the three meshes. Since the difference of PCT for all these meshes is close to zero, it indicates that an oscillatory convergence or that the “exact” solution has been attained [4.5.1]. To provide further assurance of convergence, grid convergence index (GCI), which is a measure of the solution uncertainty, is computed as 0.566%. The apparent order of the method is calculated as 1.2.

Based on the above results, it can be concluded that the Mesh 1 is reasonably converged and is adopted as the licensing basis converged mesh.

4.5.2.3 Vacuum Drying

The initial loading of SNF in the MPC requires that the water within the MPC be drained and replaced with helium. For MPC-37, MPC-37P, MPC-44, MPC-32ML and MPC-89 containing moderate burnup fuel assemblies only, this operation may be carried out using the conventional vacuum drying approach without time limits up to design basis heat load. In this method, removal of moisture from the MPC cavity is accomplished by evacuating the MPC after completion of MPC draining operation. Vacuum drying of MPC-37 , MPC-37P, MPC-44, MPC-32ML and MPC-89 containing high burnup fuel assemblies is permitted without time limit up to threshold heat loads defined in Tables 4.5.1 and 4.5.28. As described subsequently in this chapter, vacuum drying with time limits is allowed for all MPCs containing HBF greater than threshold heat loads. Table 4.5.19 provides a summary of vacuum drying conditions. High burnup fuel drying in MPCs generating greater than threshold heat load is performed by a forced flow helium drying process as discussed in Section 4.5.4.

Prior to the start of the MPC draining operation, both the HI-TRAC VW annulus and the MPC are full of water. The presence of water in the MPC ensures that the fuel cladding temperatures are lower than design basis limits by large margins. As the heat generating active fuel length is uncovered during the draining operation, the fuel and basket mass will undergo a gradual heat up from the initially cold conditions when the heated surfaces were submerged under water. To minimize fuel temperatures during vacuum drying operations the HI-TRAC VW annulus must be water filled. The necessary operational steps required to ensure this requirement are set forth in Chapter 9.

A 3-D FLUENT thermal model of the MPC is constructed in the same manner as described in Section 4.4[1]. The principal input to this model is the effective conductivity of fuel under vacuum drying operations. To bound the vacuum drying operations the effective conductivity of fuel is

1 The MPC thermal model adopted for vacuum drying analysis in this sub-section includes the gap between the intersecting basket panels as 0.4 mm. A sensitivity study of the most limiting thermal scenario (least margins to fuel temperature limit) of vacuum drying condition is performed with this gap as 0.8 mm and discussed in Sub-section 4.5.4.4.

STORM FW MPCs are also considerably thicker than their counterparts in the stainless baskets, resulting in an additional enhancement in conduction heat transfer.

4.5.4 Analysis of Limiting Thermal States During Short-Term Operations

4.5.4.1 Vacuum Drying

The vacuum drying option is evaluated for the two limiting scenarios defined in Section 4.4.1.5 to address Moderate Burnup Fuel under limiting heat load (Pattern A) and High Burnup Fuel under threshold heat load defined in Table 4.5.1 (MPC-37, MPC-44, and MPC-89) and Table 4.5.28 (MPC-32ML). The principle objective of the analysis is to ensure compliance with ISG-11 temperature limits. For this purpose 3-D FLUENT thermal models of the MPC-37~~, MPC-89~~, MPC-37P, MPC-44, and MPC-32ML canisters are constructed as described in Section 4.5.2.2 and bounding steady state temperatures computed. The results are tabulated in Tables 4.5.6, 4.5.7, 4.5.29, 4.5.30, 4.5.31, 4.5.20 and 4.5.21. The results show that the cladding temperatures comply with the ISG-11 limits for moderate and high burnup fuel in Table 4.3.1 by robust margins. The analysis presented above supports MPC drying options as summarized in Table 4.5.19.

As described in Section 4.5.2.3, a sensitivity study of the most limiting configuration under vacuum drying (i.e. MPC-37 at threshold heat load with HBF) is evaluated with fuel effective planar thermal conductivities provided in Table 4.5.8. The steady state results, which are presented in Table 4.5.27, demonstrate that the peak cladding temperature for HBF is below its ISG-11 Rev 3 temperature limit with robust margins. Since the margins to temperature limit is significantly higher for MPC with only MBF, the PCT for this condition under design basis heat loads is also below the MBF cladding temperature limit.

4.5.4.2 Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a conventional, closed loop dehumidification system consisting of a condenser, a 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. Demoisturization to the 3 torr vapor pressure criteria required by NUREG 1536 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. Appendix 2.B of [4.1.8] provides a 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 in Table 2.2.3. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode

As described in Section 4.4.1.8, the radial cold gap on the basket periphery varies with the height of the basket (see licensing drawings in Section 1.5). A series of sensitivity studies are performed for each of the MPC heights and corresponding allowable maximum radial cold gap described in the licensing drawing. The PCT and MPC cavity pressure results are presented in Table 4.5.18. The results demonstrate that the PCT is essentially the same as the results from licensing basis scenario presented in Table 4.5.2. Additionally, MPC cavity pressure is also essentially the same as licensing basis scenario presented in Table 4.5.5. Therefore, all safety conclusions made for licensing basis scenarios in this chapter remain applicable to all the scenarios defined on the licensing drawings.

An additional thermal evaluation is performed to include the ability to use water without ethylene glycol in the HI-TRAC water jacket during transfer operations below 32°F. The following methodology is used to determine the minimum decay heat such that the water jacket stays above the freezing temperature of water even when ethylene glycol is not added:

1. A steady state evaluation is performed with the minimum height MPC-37 and HI-TRAC VW.
2. Ambient temperature of 0°F (minimum ambient temperature from Table 2.2.2) and a uniform MPC decay heat defined in Table 4.5.30 is utilized.
3. The rest of the thermal modeling is identical to that presented in the preceding paragraphs of this section.

The minimum water jacket temperature from the evaluation is presented in Table 4.5.30. It is demonstrated that the water temperature is well above its freezing point, and therefore, addition of ethylene glycol is not required.

A site-specific evaluation may be performed using the site-specific heat loads, equipment models, and the ambient temperature conditions using the above methodology to establish the minimum allowable heat load for any site.

4.5.4.4 Effect of Increase in Basket Panel Gap

As described in Subsection 4.5.2.3, a sensitivity study is performed for the vacuum drying condition of high burnup fuel at threshold heat load with the basket panel notch gap equal to 0.8 mm. The results of the steady state analysis vacuum drying condition are summarized in Table 4.5.10. The PCT and cask component temperatures during vacuum drying are below their respective temperature limits. Therefore, the effect of increasing the panel gap is small and leaves sufficient safety margins during vacuum drying conditions.

4.5.4.5 Evaluation of 15x15I Short Fuel Assembly

(a) On-Site Transfer

The thermal evaluations described in this sub-section are performed for the 15x15I short fuel height of the HI-TRAC VW, similar to the evaluations performed for HI-STORM FW System in Sub-Section 4.4.1.7. The thermal model is exactly the same as that described in sub-section 4.5.2 with the following exceptions:

Normal on-site transfer using the HI-TRAC VW can be carried out inside a building. When HI-TRAC VW is located inside a building, the ambient air temperature inside the building could be higher than the outdoor environment temperature used in the thermal evaluations performed in Subsection 4.5.4.3. To evaluate this scenario, an ambient temperature that corresponds to the maximum indoor air temperature specified in Table 2.2.2 for short term operations is assumed. Since the cask is inside a building, no solar insolation is applied to the cask. A steady state analysis is performed for the limiting thermal scenario of MPC-37 inside the HI-TRAC VW under heat load pattern A. The peak cladding, MPC and the HI-TRAC component temperatures are presented in Table 4.5.9 in addition to the MPC cask cavity pressure. The predicted component temperatures and MPC cavity pressure are below their respective temperatures and pressure for outdoor environment presented in Tables 4.5.2 and 4.5.5 respectively. Therefore, the normal on-site transfer of a HI-TRAC outside the building and with solar insolation as evaluated in Subsection 4.5.4.3 is the limiting thermal condition.

4.5.5 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

NUREG-1536 requires an evaluation of cask cooldown and reflood procedures to support fuel unloading from a dry condition. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. Direct MPC cooldown is effectuated by introducing water through the lid drain line. From the drain line, water enters the MPC cavity near the MPC baseplate. Steam produced during the direct quenching process will be vented from the MPC cavity through the lid vent port. To maximize venting capacity, both vent port RVOA connections must remain open for the duration of the fuel unloading operations. As direct water quenching of hot fuel results in steam generation, it is necessary to limit the rate of water addition to avoid MPC overpressurization. For example, steam flow calculations using bounding assumptions (100% steam production and MPC at design pressure) show that the MPC is adequately protected under a reflood rate of 3715 lb/hr. Limiting the water reflood rate to this amount or less would prevent exceeding the MPC design pressure.

4.5.6 Maximum Internal Pressure (Load Case NB in Table 2.2.7)

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 VW transfer cask, the gas temperature will correspond to the thermal conditions within the MPC analyzed in Section 4.5.4.3. Based on this analysis the MPC internal pressure is computed under the assumption of maximum helium backfill specified in Table 4.4.8 and confirmed to comply with the short term operations pressure limit in Table 2.2.1. The results are tabulated in Table 4.5.5.

4.5.7 HI-DRIP

The design function of HI-DRIP is to protect the water in the MPC contained in the transfer cask from boiling during the interval after it has been lifted out of the fuel pool and is subject to surface decontamination, lid welding, and related operations which precede the evacuation of water from the MPC followed by drying. As such, HI-DRIP is essentially a device to prevent boiling of the MPC water during the period it is full of water due to the decay heat produced by the MPC contents extending its time-to-boil (TTB) indefinitely. The HI-DRIP is system has been approved for use with HI-STORM 100 system ~~The HI-DRIP is system has been approved for use with HI-STORM 100 system [$]~~[4.1.8].

The operations needed to package the fuel in an MPC (hereafter called "MPC packaging operations") can be divided into three intervals; namely, (1) When the MPC is full of water and in the pool, (2) When the MPC, full of water, is out of the pool in an ambient environment, and (3) The bulk water from the MPC is pumped out and its contents are dried by the FGD eliminating the risk of uncontrolled pressure rise from rising water vapor pressure even if the MPC were to be sealed shut. Interval 1 is not of concern because submergence in the pool's large inventory of water (typically maintained at below 120°F through the fuel pool cooling system) precluded the potential of boiling. Interval 3 is likewise immune to boiling because in this interval, water has been removed from the system. Interval 2, however, is vulnerable to uncontrolled boiling of water because the heat rejection rate to the ambient air is quite low. If the canister's heat generation rate, Q, is sufficiently high (typically over 20kW) then the heat dissipation rate from the cask to the ambient environment cannot keep pace with the decay heat generation rate causing the canister to heat up monotonically, eventually leading its contained water to reach the boiling point. Boiling of water in the canister should be avoided to prevent excessive water vapor in the vicinity of the lid-to-shell weld puddle (which may degrade the weld quality) and to prevent uncontrolled loss of shielding water. Because the MPC internal space is vented to the environment throughout the duration of Interval 2, there is no risk of vapor over-pressure. It serves a valuable ALARA function by eliminating TTB limitations and the associated human activities that accrete dose to the crew.
HI-DRIP consists of a ring that girdles the transfer cask around its main cylindrical body. The ring is equipped with small spray nozzles uniformly spaced around its circumference such that the entire circumference of the transfer cask shell can be drenched by the spray from them. The ring is connected to the plant's water supply with a gate valve serving to regulate the flow of water to the ring. The amount of water sprayed should be set such that the entire surface of the cask is wetted by gravity. As discussed below, as long as the external surface of the transfer cask remains wet, the water in the canister will remain below boiling. It is noted that the operation of HI-DRIP does not require any pump or electric power; the motive pressure is provided entirely by the plant's water supply system. (In PWR pools, draining un-borated water into the pool would reduce its boric acid concentration. This fact must be considered in setting up HI-DRIP if the pool is used as the recipient of the drain).

The difference between the MPC water and the surface temperature of a HI-TRAC class of transfer casks containing MPC and its annulus filled with water arrayed vertically in an ambient

environment can be obtained by simply calculating the total thermal resistance between the HI-TRAC surface and MPC water. However, a more detailed CFD analysis was performed and found to have this temperature difference less than 45°F less than the temperature of the boiling water (i.e., 212°F). It can be readily deduced by intuitive reasoning that at a lower MPC water temperature, the difference will be smaller. Thus, the temperature of the HI-TRAC surface (say, approximately at its mid-height) will be greater than 165°F when the MPC water is boiling. A scoping calculation to illustrate that HI-DRIP will not permit the MPC water to boil is provided below for a typical application:

Assuming that the cask surface temp is 165°F, the ambient air is at 110°F, the heat rejection rate from the shortest and smallest HI-TRAC i.e. 7.5 feet dia by 14 feet high cask surface wetted by water, is conservatively estimated to be, Z = (15)(55)(3.14)(7.5)(15)=291431 BTU/hr or 85kW, where the heat transfer coefficient from the vaporization-aided cask surface is conservatively assumed to be 15 BTU/ft^2-hr-°F. Therefore, when the MPC water is boiling, the external surface of the cask will reject approximately 85 kW, in excess of a typical maximum cask heat load of approximately 43 kW. Therefore, it follows that a cask kept wet by HI-DRIP will never retain sufficient quantity of heat to cause boiling of canister water. In other words, the TTB will be infinity if HI-DRIP is used.

Application:
1) The following preparatory calculations shall be performed during the design phase of this ancillary:

   a. Using the NRC reviewed FLUENT model, compute the average surface temperature (T') at the mid-height of the cask for applicable heat load, ambient temperature and cooling water temperature assuming continuous wetting.
   b. Compute the mid-height surface temperature as in (a) above, except assume that the cask is dry surrounded by ambient air (T).
   c. Use the FLUENT solution to refine the estimate of cooling water flow rate required.
   d. Use the simplified adiabatic heat up method described in the FSAR to compute the "very conservative time to boil (t').

2) HI-DRIP should be mounted in the neck region of the transfer cask and connected to the plant's water supply. The gate valve should be adjusted such that the spray nozzles keep the cask cylindrical surface wet continuously all around its cylindrical surface.
3) The cooling water drip should be initiated no later than 50% of t' into Interval 2.

TABLE 4.5.1
THRESHOLD HEAT LOADS UNDER VACUUM DRYING OF HIGH BURNUP FUEL (See Figures 1.2.1a and 1.2.2)

MPC-37			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 34.36			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.80	9	7.2
2	0.97	12	11.64
3	0.97	16	15.52
MPC-44			
Number of Regions:1			
Number of Storage Cells: 44			
Maximum Heat Load: 30kW			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.681	44	30.0
MPC-89			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Heat Load: 34.75			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.35	9	3.15
2	0.35	40	14.00
3	0.44	40	17.60

MPC-37P			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 33.3			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.900	9	8.1
2	0.900	12	10.8
3	0.900	16	~~14.4~~14.4

Notes:s

(1) The maximum per storage cell heat load and total threshold heat load documented herein are used to perform thermal evaluations documented in subsection 4.5.4.1.

(2) The maximum per storage cell heat load and total threshold heat load allowed for an MPC-37 per the CoC are 0.8 kW and 29.6 kW respectively. Similarly, these values for an MPC-89 per the CoC are 0.337 kW and 30 kW respectively. No additional thermal analysis is performed for these lower allowable CoC heat loads since they are bounded by the heat loads used in the vacuum drying thermal calculations (subsection 4.5.4.1).

Table 4.5.8

EFFECTIVE CONDUCTIVITY OF DESIGN BASIS FUEL[Note 1] UNDER VACUUM DRYING OPERATIONS
(Btu/hr-ft-°F)

Ft. Calhoun 14x14[Note 1]		
Temperature (°F)	Planar	Axial
200	0.125	0.726
450	0.265	0.793
700	0.498	0.886
	0.623@800°F	1.024@1000°F
Note 1: Ft. Calhoun 14x14 fuel is defined as the design basis fuel under the limiting condition of fuel storage in the minimum height MPC-37 (See Table 4.4.2).		
16x16D[Note 1]		
Temperature (°F)	Planar	Axial
212	0.095	0.8
450	0.229	0.867
700	0.458	0.962
785	0.558	1.003
14x14D[Note 2]		
Temperature (°F)	Planar	Axial
200	0.101	0.675
450	0.231	0.738
700	0.444	0.825
1000	0.817	0.954
15x15I in MPC-37P		
Temperature (°F)	Planar	Axial
200	0.116	0.706
450	0.239	0.771
700	0.441	0.862
	0.548@800F 0.813@1000F	0.996@1000F

BWR Fuel		
Temperature (ºF)	Planar	Axial
200	0.112	1.004
450	0.236	1.095
700	0.441	1.221
	0.553@800ºF	1.408@1000ºF
Note 1: Design Basis MPC-32ML fuel. Note 2: Design Basis MPC-44 Fuel.		

Table 4.5.19
MPC DRYING OPERATIONS

MPC Type	Fuel	Heat Load Limit (kW)	Method of Drying
MPC-32ML	MBF	44.16 (Note 1)	FHD/Vacuum Drying without Time Limit
	HBF	44.16 (Note 1)	FHD/Vacuum Drying with Time Limit
		28.704	FHD/Vacuum Drying without Time Limit
MPC-37	MBF	44.09 (Pattern A) 45.0 (Pattern B) 37.4/39.95/44.85 (Figures 1.2.3a, 1.2.4a, 1.2.5a) 34.4 (Figures 1.2.3b/c) 36.65 (Figures 1.2.4b/c) 40.95 (Figures 1.2.5b/c) (Note 1)	FHD/Vacuum Drying without Time Limit
	HBF	44.09 (Pattern A) 45.0 (Pattern B) (Note 1)	FHD/Vacuum Drying with Time Limit
		29.6	FHD/Vacuum Drying without Time Limit
MPC-89	MBF	46.36 (Table 1.2.4a) 46.2 (Figure 1.2.6a) 44.92 (Figure 1.2.6b) 46.14 (Figure 1.2.7a) 44.98 (Figure 1.2.7b) (Note 1)	FHD/Vacuum Drying without Time Limit
	HBF	46.36 (Note 1)	FHD/Vacuum Drying with Time Limit
		30	FHD/Vacuum Drying without Time Limit
MPC-37P	MBF~~MBF~~	45.0 (Table 1.2.3c)~~$~~ 44.09 (Pattern A) 45.0 (Pattern B) (Note 1)	FHD/Vacuum Drying without Time Limit~~$~~
	HBF~~HBF~~	45.0 (Table 1.2.3c) 44.09 (Pattern A) 45.0 (Pattern B) (Note 1)~~$~~	FHD/Vacuum Drying with Time Limit~~$~~
		33.3	FHD/Vacuum Drying without Time Limit

MPC-44	MBF~~MBF~~	44.0 (Table 1.2.3e) (Note 1)~~44.0 (Table 1.2.3e) (Note 1)~~	FHD/Vacuum Drying without Time Limit~~FHD/Vacuum Drying without Time Limit~~
	HBF~~HBF~~	44.0 (Table 1.2.3e) (Note 1)~~30.0~~	FHD/Vacuum Drying with Time Limit~~FHD/Vacuum Drying without Time Limit~~
		30.0	FHD/Vacuum Drying without Time Limit

Note 1: Design Basis heat load. Note 2: Cyclic drying under time limited vacuum drying operations is permitted in accordance with ISG-11, Rev. 3 requirements by limiting number of cycles to less than 10 and cladding temperature variations to less than 65ºC (117ºF). Suitable time limits for these cycles shall be evaluated based on site specific conditions and thermal methodology defined in Section 4.5.

Table 4.5.29

MAXIMUM COMPONENT TEMPE~~A~~RATURES UNDER VACUUM DRYING OPERATIONS OF MPC-32ML

Component	Temperature @ Threshold Heat (HBF) °C (°F)	Temperature @ Design Maximum Heat (MBF) °C (°F)
Fuel Cladding	384 (723)	481 (898)
MPC Basket	368 (694)	461 (862)
Basket Periphery	304 (579)	369 (696)
Aluminum Basket Shims	263 (505)	314 (597)
MPC Shell	160 (320)	178 (352)
MPC Lid[1]	100 (212)	102 (216)

1 Maximum section average temperature is reported.

Table 4.5.30
MAXIMUM COMPONENT TEMPERATURES UNDER LOW AMBIENT TEMPERATURE CONDITIONS

Heat Load	**Minimum Water Temperature, ^{o}F**
12 kW	50

Table 4.5.31 MAXIMUM COMPONENT TEMPERATURES UNDER VACUUM DRYING OPERATIONS OF MPC-37P		
Component	**Temperature @ Threshold Heat (HBF) °C (°F)**	**Temperature @ Design Maximum Heat (MBF) °C (°F)**
Fuel Cladding	379 (714)	473 (883)
MPC Basket	356 (673)	425 (797)
Aluminum Basket Shims	256 (493)	314 (597)
MPC Shell	145 (293)	160 (320)
MPC Lid[1]	136 (277)	155 (311)

Table 4.5.32 MAXIMUM COMPONENT TEMPERATURES UNDER VACUUM DRYING OPERATIONS OF MPC-44		
Component	**Temperature @ Threshold Heat (HBF) °C (°F)**	**Temperature @ Design Maximum Heat (MBF) °C (°F)**
Fuel Cladding	444 (831)	368 (694)
MPC Basket	430 (806)	354 (669)
Aluminum Basket Shims	302 (576)	258 (496)
MPC Shell	237 (459)	210 (410)
MPC Lid[1]	104 (219)	99 (507)

1 Maximum section average temperature is reported.

4.6 OFF-NORMAL AND ACCIDENT EVENTS

In this Section thermal evaluation of HI-STORM FW System under off-normal and accident conditions defined in Sections 4.6.1 and 4.6.2 is provided. To ensure a bounding evaluation the limiting Pattern A thermal loading scenario with MPC-37 defined in Section 4.4.4 is adopted in the evaluation.

4.6.1 Off-Normal Events

4.6.1.1 Off-Normal Pressure (Load Case NB in Table 2.2.7)

This event is defined as a combination of (a) maximum helium backfill pressure (Table 4.4.8), (b) 10% fuel rods rupture, (c) limiting fuel storage configuration and (d) off-normal ambient temperature. 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 4.6.7. The result is 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 results of this event (maximum temperatures and pressures) are provided in Table 4.6.1 and 4.6.7. The results are below the off-normal condition temperature and pressure limits (Tables 2.2.3 and 2.2.1).

4.6.1.3 Partial Blockage of Air Inlets/Outlets

The HI-STORM FW 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 the HI-STORM FW air inlet vents and/or outlet vents are 50% blocked. The resulting decrease in flow area increases the flow resistance of the inlet and outlet ducts, thereby decreasing the air mass flow ate into the system.

An explicit thermal evaluation to evaluate the effect of 50% blockage of air inlet vents is performed. The effect of the increased flow resistance on fuel and other component 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 4.6.7. 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.

In an unlikely event of both inlet and outlet vents being 50% blocked, cold air still enters into the annulus space between the MPC and HI-STORM FW overpack and hot air exits from the partially unblocked outlet vents. The effect of partially blocked outlet vents is similar to the

The results of the fire and post-fire events are reported in Table 4.6.2. These results demonstrate that the fire accident event has a minor effect on the fuel cladding temperature. In addition, the local concrete temperature is well below its short-term temperature limit (Table 2.2.3). The temperatures of the basket and components of MPC and HI-STORM FW overpack (see Table 4.6.2) are within the allowable temperature limits.

Table 4.6.2 shows a slight increase in fuel temperature following the fire event. Thus the impact on the MPC internal helium pressure is correspondingly small. Based on a conservative analysis of the HI-STORM FW 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. Thus, the ability of the HI-STORM FW system to maintain the spent nuclear fuel within design temperature limits during and after fire is assured.

(b) HI-TRAC VW 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 VW is initially (before fire) assumed to be loaded to 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, as discussed in Subsection 4.6.2.1(a) is assumed to surround the HI-TRAC VW in a 1 m wide ring. The fire parameters are same as that assumed for the HI-STORM FW fire discussed in this preceding subsection. In this analysis, the HI-TRAC VW and its contents are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat and heat input from the fire.

Based on the specified 50 gallon fuel volume, HI-TRAC VW cylinder diameter (7.9 ft) and the 1 m fuel ring width, the fuel ring area is 115.2 ft^2 and has a depth of 0.696 in. From this depth and the fuel consumption rate of 0.15 in/min, the fire duration τ_f is calculated to be 4.64 minutes (279 seconds). The fuel consumption rate of 0.15 in/min is a lowerbound value from Sandia Report [4.6.1]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

From the HI-TRAC VW fire analysis, a bounding rate of temperature rise 2.722°F per minute is determined. Therefore, the total temperature rise is computed as the product of the rate of temperature rise and τ_f is 12.6°F. Because the cladding temperature at the start of fire is substantially below the accident temperature limit, the fuel cladding temperature limit during HI-TRAC VW fire is not exceeded. To confirm that the MPC pressure remains below the design accident pressure (Table 2.2.1) the MPC pressure resulting from fire temperature rise is computed using the Ideal Gas Law. The result (see Table 4.6.7) is below the pressure limit (see Table 2.2.1).

An alternate method using the FLUENT thermal model described in Section 4.5 can be adopted to evaluate HI-TRAC site-specific fire accident event. Principal modeling steps and acceptance criteria are defined in Table 4.6.11. This approach is consistent with that approved in HI-STORM 100 FSAR [4.1.8].

4.6.2.2 Jacket Water Loss

In this subsection, the fuel cladding and MPC boundary integrity is evaluated under a postulated (non-mechanistic) loss of water from the HI-TRAC VW water jacket. For a bounding analysis, all water compartments are assumed to lose their water and be replaced with air. The HI-TRAC VW 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 VW 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 4.6.7) shows a positive margin of safety.

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 FW system to reach steady state conditions. Because of the large mass of the HI-STORM FW 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 FW system are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures (45°F). The HI-STORM FW extreme ambient temperatures computed in this manner are reported in Table 4.6.4. The co-incident MPC pressure is also computed (Table 4.6.7) and compared with the accident design pressure (Table 2.2.1), which shows a positive safety margin. 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 eight bottom inlets for a significant duration (32 hours). 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

lowerbound on □□, the HI-STORM FW 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 □□ is computed as follows:

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

where:

- $\Delta\tau$ = minimum available burial time (hr)
- m = Mass of HI-STORM FW 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, the minimum available burial time is computed as 57.6 hours for the short fuel assembly (15x15I). A site-specific calculation based on the methodology described herein can be performed to determine the burial time limits. The co-incident MPC pressure (see Table 4.6.7) is also computed and compared with the accident design pressure (Table 2.2.1). These results indicate that HI-STORM FW has a substantial thermal sink capacity to withstand complete burial-under-debris events.

An explicit site-specific evaluation may also be performed using CFD to compute the allowable burial time utilizing the site-specific conditions. The following methodology shall be applied to perform the CFD analysis:

[

Withheld in accordance with 10 CFR 2.390

]

The steps for the site-specific evaluations are presented in Table 4.6.12.

4.6.2.6 Evaluation of Smart Flood (Load Case AD in Table 2.2.13)

A number of design measures are taken in the HI-STORM FW system to limit the fuel cladding temperature rise under a most adverse flood event (i.e., one that is just high enough to block the inlet duct). An unlikely adverse flood accident is assumed to occur with flood water upto the inlet height and is termed as 'smart flood'. The inlet duct is narrow and tall so that blocking the inlet ducts completely would require that flood waters wet the bottom region of the MPC creating a heat sink.

The inlet duct is configured to block radiation efficiently even if the radiation emanating from

Table 4.6.11 PRINCIPAL SITE-SPECIFIC HI-TRAC FIRE ACCIDENT MODELING STEPS
[Withheld in accordance with 10 CFR 2.390]

Table 4.6.12 PRINCIPAL SITE-SPECIFIC HI-STORM BURIAL UNDER DEBRIS ACCIDENT MODELING STEPS
~~:~~[Withheld in accordance with 10 CFR 2.390]

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CHAPTER 5†: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM FW system is presented in this chapter. As described in Chapter 1, the HI-STORM FW system is designed to accommodate both PWR and BWR MPCs within HI-STORM FW overpacks (see Table 1.0.1).

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM FW system is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 2.1. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

As described in Chapter 2 (see Table 2.1.1), MPC-37 and MPC-32ML are designed to store various PWR fuel assembly classes. In this chapter, shielding analyses are mainly performed for MPC-37 (for PWR fuel assemblies) since most PWR fuel assemblies can be loaded into MPC-37. Nevertheless, shielding analysis for adjacent and 1-m dose rates for HI-STORM FW cask with MPC-32ML loaded with 16x16D fuel assemblies are included in this chapter, to show the radiation shielding features of the cask system is enough to meet the requirements of 10CFR72.104 and 72.106. Site specific analyses need to use the site specific MPC and fuel type for controlled area boundary dose calculations to show the site's compliance with 10 CFR 72.104. Also, as discussed in Section 5.1, the burnup and cooling times selected for accident conditions represent reasonable upper bound limit, and the heavy metal mass in MPC-32ML is comparable to that in MPC-37. This is confirmed by source terms for accident-condition design basis fuel burnup, cooling time combination, as provided in Section 5.2 for fuel assembly in MPC-37 and MPC-32ML. Therefore, it is concluded that the accident condition evaluated in this chapter for MPC-37 is reasonably conservative, and no further site's compliance with 10 CFR 72.106 is required for MPC-32ML, except for the fuel assemblies in MPC-32ML with a higher burnup than the design basis accident-condition burnup. More detail is provided in Subsection 5.1.2.

Shielding analysis for adjacent and 1-m dose rates for HI-STORM FW Version E cask with MPC-44 loaded with Westinghouse 14x14 fuel assemblies are included in this chapter, to show the radiation shielding features of the cask system is enough to meet the requirements of 10CFR72.104 and 72.106. Site specific analyses need to use the site specific MPC and fuel type for controlled area boundary dose calculations to show the site's compliance with 10 CFR 72.104.

† 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 and component nomenclature of the Bill-of-Materials (Section 1.5).

CHAPTER 5[†]: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM FW system is presented in this chapter. As described in Chapter 1, the HI-STORM FW system is designed to accommodate both PWR and BWR MPCs within HI-STORM FW overpacks (see Table 1.0.1).

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM FW system is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 2.1. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

As described in Chapter 2 (see Table 2.1.1), MPC-37 and MPC-32ML are designed to store various PWR fuel assembly classes. In this chapter, shielding analyses are mainly performed for MPC-37 (for PWR fuel assemblies) since most PWR fuel assemblies can be loaded into MPC-37. Nevertheless, shielding analysis for adjacent and 1-m dose rates for HI-STORM FW cask with MPC-32ML loaded with 16x16D fuel assemblies are included in this chapter, to show the radiation shielding features of the cask system is enough to meet the requirements of 10CFR72.104 and 72.106. Site specific analyses need to use the site specific MPC and fuel type for controlled area boundary dose calculations to show the site's compliance with 10 CFR 72.104. Also, as discussed in Section 5.1, the burnup and cooling times selected for accident conditions represent reasonable upper bound limit, and the heavy metal mass in MPC-32ML is comparable to that in MPC-37. This is confirmed by source terms for accident-condition design basis fuel burnup, cooling time combination, as provided in Section 5.2 for fuel assembly in MPC-37 and MPC-32ML. Therefore, it is concluded that the accident condition evaluated in this chapter for MPC-37 is reasonably conservative, and no further site's compliance with 10 CFR 72.106 is required for MPC-32ML, except for the fuel assemblies in MPC-32ML with a higher burnup than the design basis accident-condition burnup. More detail is provided in Subsection 5.1.2.

Shielding analysis for adjacent and 1-m dose rates for HI-STORM FW Version E cask with MPC-44 loaded with Westinghouse 14x14 fuel assemblies are included in this chapter, to show the radiation shielding features of the cask system is enough to meet the requirements of 10CFR72.104 and 72.106. Site specific analyses need to use the site specific MPC and fuel type for controlled area boundary dose calculations to show the site's compliance with 10 CFR 72.104.

[†] 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 and component nomenclature of the Bill-of-Materials (Section 1.5).

10CFR72 contains two sections that set down main dose rate requirements: §104 for normal and off-normal conditions, and §106 for accident conditions. The relationship of these requirements to the analyses in this Chapter 5, and the burnup and cooling times selected for the various analyses, are as follows:

- 10CFR72.104 specifies the dose limits from an ISFSI (and other operations) at a site boundary under normal and off-normal conditions. Compliance with §104 can therefore only be demonstrated on a site-specific basis, since it depends not only on the design of the cask system and the loaded fuel, but also on the ISFSI layout, the distance to the site boundary, and possibly other factors such as use of higher density concrete or the terrain around the ISFSI. The purpose of this chapter is therefore to present a general overview over the expected or maximum dose rates, next to the casks and at various distances, to aid the user in applying ALARA considerations and planning of the ISFSI.
- For the accident dose limit in 10CFR72.106 it is desirable to show compliance in this Chapter 5 on a generic basis, so that calculations on a site-by-site basis are not required†. To that extent, a burnup and cooling time calculation that maximizes the dose rate under accident conditions needs to be selected.

It is recognized that for a given heat load, an infinite number of burnup and cooling time combination could be selected, which would result in slightly different dose rate distributions around the cask. For a high burnup with a corresponding longer cooling time, dose locations with a high neutron contribution would show higher dose values, due to the non-linear relationship between burnup and neutron source term. At other locations dose rates are more dominated by contribution from the gamma sources. In these cases, short cooling time and lower burnup combinations with heat load comparable to the higher burnup and corresponding longer cooling time combinations would result in higher dose rates. However, in those cases, there would always be a compensatory effect, since for each dose location, higher neutron dose rates would be partly offset by lower gamma dose rates and vice versa. This is further complicated by the regionalized loading patterns qualified from a thermal perspective and shown in Figure 1.2.3 through Figure 1.2.5 for MPC-37 and Figures 1.2.6 and 1.2.7 for MPC-89. These contain cells with substantially different heat load limits, and hence substantially different ranges of burnup, enrichment and cooling time combinations. The approach to cover all those variations in a conservative way is outlined below.

To prescribe radiological limits for the fuel to be loaded, loading curves are defined in Tables 2.1.9 and 2.1.10, where a loading curve specifies the minimum cooling time as a function of fuel burnup. Different loading curves are defined for the different heat load limits, so that the thermal and radiological requirements for the fuel in each cell are approximately aligned. However, it should be noted that thermal and radiological limits for each assembly are applied completely independent from each other. The uniform and regionalized loading curves for the fuel to be loaded in the MPC-37, MPC-32ML, MPC-37P, MPC-44 or MPC-89 canisters are discussed in Subsection 5.2.7.

† As it is discussed in Subsection 5.1.2, a site-specific shielding evaluation may be required for accident-condition of MPC-32ML.

To determine dose rates consistent with both the uniform and regionalized thermal loading, it is necessary to consider the ranges of burnup and cooling times from all loading curves. For that, 8 burnup values between 5 and 70 GWd/mtU are selected, and corresponding minimum required cooling times are established and used in the dose analyzes. The heat load patterns in Figures 1.2.3 through 1.2.7 contain from 5 to 20 regions each, ~~i.e.~~i.e., from 5 to 20 principal locations with different heat load limit. Applying 8 burnup and cooling time combinations to each location would result up to 8^{20} = 1.15E+18 different burnup and cooling time loading arrangements per pattern. Analyzing and comparing those many arrangements would be excessive. Therefore, for the radiological evaluations, some regions and loading patterns (MPC-37) are combined using the highest heat load limit (source term) of each group. For MPC-37, the heat loads for each cell are based on the "Long" fuel heat loads in Figure 1.2.5a. ~~The established bounding heat load limits are provided in Tables 5.0.3 and 5.0.4.~~

This then results in effectively only 2 or 5 regions to be independently varied for the considered bounding MPC-37 and MPC-89 patterns, and hence 8^2 = 64 or 8^5 = 32,768 different burnup and cooling time arrangements per pattern is to be analyzed, which is manageable. The selected burnup, enrichment and cooling time combinations for the uniform and regionalized loading patterns are listed in Tables 5.0.3, 5.0.4a, 5.0.4b, ~~and~~ 5.0.5, 5.0.6 and 5.0.7. ~~-~~The dose rates in the various important locations are calculated for each of these combination arrangements and the maximum is determined for each dose rate location. It should be noted that this maximum can be from a different loading arrangement in different locations.

Based on this approach, the source terms used in the analyses of MPC-37, MPC-32ML, MPC-37P, MPC-44 or MPC-89 are reasonably bounding for all realistically expected assemblies. All dose rates in this chapter are developed using this approach, unless noted otherwise. Also, as discussed in Section 5.2, the design basis BPRA activities are considered for MPC-37, MPC-32ML and MPC-44~~32ML~~ in this chapter, unless noted otherwise.

All dose rates in Section 5.1 are developed using the approach discussed above. Some dose rates in Section 5.4 were retained from previous versions of the FSAR and that are based on a representative (while still conservative) uniform loading pattern, as discussed in that Section.

Table 5.0.3

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-37 LOADING PATTERNS BASED ON FIGURES 1.2.3 THROUGH 1.2.5 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)
High Heat Load Basket Regions	5000	1.1	1.0
	10000	1.1	1.0
	20000	1.6	1.0
	30000	2.4	1.4
	40000	3.0	1.6
	50000	3.6	2.0
	60000	3.9	2.2
	70000	4.5	2.8
Low Heat Load Basket Regions	5000	1.1	1.4
	10000	1.1	2.0
	20000	1.6	3.0
	30000	2.4	4.0
	40000	3.0	6.0
	50000	3.6	10.0
	60000	3.9	18.0
	70000	4.5	29.0

~~NOTE:~~

~~To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.3 through 1.2.5.~~

Table 5.0.4a

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.6 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)
High Heat Load Basket Regions	5000	0.7	1.0
	10000	0.9	1.0
	20000	1.6	1.0
	30000	2.4	1.0
	40000	3.0	1.2
	50000	3.3	1.6
	60000	3.7	1.8
	70000	4.0	2.4
Low Heat Load Basket Regions	5000	0.7	1.4
	10000	0.9	2.0
	20000	1.6	3.0
	30000	2.4	4.0
	40000	3.0	6.0
	50000	3.3	10.0
	60000	3.7	18.0
	70000	4.0	29.0

~~NOTE:~~

~~To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.6.~~

Table 5.0.4b (continued)

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.7 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)
Low Heat Load Basket Regions (Region 4)	5000	0.7	1.0
	10000	0.9	1.4
	20000	1.6	2.2
	30000	2.4	2.8
	40000	3.0	3.5
	50000	3.3	5.0
	60000	3.7	7.0
	70000	4.0	9.0
Low Heat Load Basket Regions (Region 5)	5000	0.7	1.4
	10000	0.9	2.0
	20000	1.6	3.0
	30000	2.4	4.0
	40000	3.0	6.0
	50000	3.3	10.0
	60000	3.7	18.0
	70000	4.0	29.0

~~NOTE:~~

~~To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.7.~~

Table 5.0.5

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-32ML LOADING PATTERNS FOR NORMAL CONDITIONS

Burnup (MWD/MTU)	Initial U-235 Enrichment (wt%)	Cooling Time (years)
15000	1.1	3
20000	1.1	3
25000	1.6	3.5
30000	2	3.6
35000	2.4	4
40000	2.6	4.5
45000	3	5
50000	3.3	6
55000	3.6	7
60000	3.6	9
65000	3.9	11
70000	4.2	13

Table 5.0.6

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-44 UNIFORM LOADING PATTERNS FOR NORMAL CONDITIONS

Burnup (MWD/MTU)	Initials U-235 Enrichment (wt%)	Cooling Time (years)
5000	1.1	3.0
10000	1.1	3.0
20000	1.6	3.0
30000	2.4	3.5
40000	3.0	5.0
50000	3.6	7.0
60000	3.9	11.0
70000	4.5	21.0

Table 5.0.7

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-37P LOADING PATTERNS FOR NORMAL CONDITIONS

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)
Low Heat Load Basket Region (Region 1)	5000	1.1	1.6
	10000	1.1	1.6
	20000	1.6	3.5
	30000	2.4	4.5
	40000	3.0	6.0
	50000	3.6	12.0
	60000	3.9	21.0
	70000	4.5	33.0
High Heat Load Basket Region (Region 2)	5000	1.1	1.6
	10000	1.1	1.6
	20000	1.6	1.6
	30000	2.4	2.0
	40000	3.0	2.2
	50000	3.6	2.6
	60000	3.9	3.0
	70000	4.5	3.5
High Heat Load Basket Region (Region 3)	5000	1.1	1.6
	10000	1.1	1.6
	20000	1.6	1.6
	30000	2.4	1.6
	40000	3.0	1.6
	50000	3.6	1.8
	60000	3.9	2.4
	70000	4.5	3.0

Table 5.0.7 (continued)

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-37P LOADING PATTERNS FOR NORMAL CONDITIONS

Medium Heat Load Basket Region (Region 4)	5000	1.1	1.6
	10000	1.1	1.6
	20000	1.6	1.6
	30000	2.4	2.8
	40000	3.0	3.5
	50000	3.6	5.0
	60000	3.9	7.0
	70000	4.5	9.0
Medium Heat Load Basket Region (Region 5)	5000	1.1	1.6
	10000	1.1	1.6
	20000	1.6	2.8
	30000	2.4	4.0
	40000	3.0	5.0
	50000	3.6	8.0
	60000	3.9	13.0
	70000	4.5	24.0

Table~~s~~ 5.1.10 provides dose rates adjacent to and one meter from the HI-TRAC VW Version V2 during normal conditions for the MPC-89. The dose rates listed in Table 5.1.10 correspond to the normal condition in which the MPC is dry and the Gamma Shield Cylinder and Neutron Shield Cylinder are present.

Tables 5.1.5, 5.1.6 and 5.1.11 provide the design basis dose rates adjacent to the HI-STORM FW overpack during normal conditions for the MPC-37, MPC-89 and MPC-32ML. Tables 5.1.7, 5.1.8 and 5.1.12 provide the design basis dose rates at one meter from the HI-STORM FW overpack containing the MPC-37, MPC-89 and MPC-32ML, respectively.

Table 5.1.14 provides the design basis dose rates adjacent to the HI-STORM FW Version E overpack during normal conditions for the MPC-44. Table 5.1.15 provides the design basis dose rates at one meter from the HI-STORM FW Version E overpack containing the MPC-44. Table 5.1.16 provides dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-44.

Table 5.1.17 provides the design basis dose rates adjacent to the HI-STORM FW Version E overpack during normal conditions for the MPC-37P. Table 5.1.18 provides the design basis dose rates at one meter from the HI-STORM FW Version E overpack containing the MPC-37P. Table 5.1.19 provides dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37P.

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. Table 5.1.3 presents the annual dose to an individual from a single HI-STORM FW 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. 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. 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 FW cask array configurations provided in Table 5.1.3. Figure 5.1.4 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.4.21. These curves, which are based on an 8760 hour occupancy, are provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Subsection 5.2.3 discusses the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM FW system. Subsection 5.4.4 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

The analyses summarized in this section demonstrate that the HI-STORM FW system is in compliance with the radiation and exposure objectives of 10CFR72.106. Since only representative dose rate values for normal conditions are presented in this chapter, compliance

with 10CFR72.104 is not being evaluated. This will be performed as part of the site specific evaluations.

5.1.2 Accident Conditions

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

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

Structural evaluations, presented in Chapter 3, shows that a freestanding HI-STORM FW storage overpack containing a loaded MPC remains standing during events that could potentially lead to a tip-over event. Therefore, the tip-over accident is not considered as part of the shielding evaluation.

Design basis accidents which may affect the HI-STORM FW 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 doses for the loaded HI-STORM FW overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits. ~~However~~However, the adjacent and one meter dose rates may be increased, which should be considered in any post-accident activities near the affected cask.

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 MPC’s design features (see Chapter 1). Further, the structural evaluation of the HI-TRAC VW in Chapter 3 shows that the inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell is possible; however, localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the

Table 5.1.14

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW VERSION E OVERPACK
FOR NORMAL CONDITIONS
MPC-44 WITH 14X14 FUEL
UNIFORM LOADING PATTERNS (SEE TABLE 5.0.6)

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)
1	316.6	0.9	34.9	6.1	358.5	454.5
2	177.9	0.7	0.3	0.7	179.6	232.3
3 (surface)	71.8	0.2	59.3	2.8	134.1	230.8
3 (overpack edge)	32.1	0.1	26.8	1.3	60.4	103.8
4 (center)	0.1	2.2	0.6	0.7	3.6	5.4
4 (mid)	<0.1	1.9	0.6	0.6	3.2	4.9
4 (outer)	0.5	<0.1	0.6	<0.1	1.1	2.0

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values (in units of mrem/hr) are rounded to the nearest tenths place.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

Table 5.1.15

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW VERSION E OVERPACK
FOR NORMAL CONDITIONS
MPC-44 WITH 14X14 FUEL
UNIFORM LOADING PATTERNS (SEE TABLE 5.0.6)

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)
1	91.3	0.3	3.3	0.5	95.4	122.5
2	107.3	0.3	2.0	0.4	110.1	147.9
3	15.8	<0.1	7.7	0.3	23.9	38.6
4 (center)	1.9	<0.1	1.2	<0.1	3.2	5.5

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values (in units of mrem/hr) are rounded to the nearest tenths place.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

Table 5.1.16

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-44 WITH 14X14 FUEL
UNIFORM LOADING PATTERNS (SEE TABLE 5.0.6)

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 HI-TRAC VW						
1	1203.0	11.6	961.4	30.9	2206.9	2701.8
2	3525.5	34.6	<0.1	70.5	3630.5	5111.2
3	36.1	2.8	567.7	3.4	610.1	1446.8
4	73.3	2.1	583.3	357.8	1016.5	2100.0
5	269.0	5.7	1309.4	2530.9	4115.0	4364.0
ONE METER FROM THE HI-TRAC VW						
1	712.0	5.4	122.3	14.5	854.2	1143.8
2	1734.6	10.3	12.8	25.1	1782.8	2546.1
3	232.4	2.4	161.5	3.8	400.2	735.2
4	89.3	0.2	372.6	36.2	498.2	1050.1
5	634.3	0.4	1267.3	139.5	2041.4	2273.8

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values (in units of mrem/hr) are rounded to the nearest tenths place.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

Table 5.1.17

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW VERSION E OVERPACK
FOR NORMAL CONDITIONS
MPC-37P WITH CE15X15 FUEL
LOADING PATTERNS (SEE TABLE 5.0.7)

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)**
1	459.4	0.6	9.9	2.9	472.7	534.2
2	318.7	0.3	<0.1	0.2	319.3	356.7
3 (surface)	58.9	0.7	48.3	8.1	116.0	187.2
3 (overpack edge)	26.6	0.3	21.5	4.0	52.4	83.9
4 (center)	0.6	2.0	0.8	0.6	4.0	5.1
4 (mid)	0.7	1.6	0.8	0.5	3.5	4.7
4 (outer)	0.9	<0.1	0.4	<0.1	1.3	1.9

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values (in units of mrem/hr) are rounded to the nearest tenths place.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

Table 5.1.18

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW VERSION E OVERPACK
FOR NORMAL CONDITIONS
MPC-37P WITH CE15X15 FUEL
LOADING PATTERNS (SEE TABLE 5.0.7)

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)
1	150.4	0.1	2.3	0.2	153.0	172.2
2	166.6	0.1	1.6	0.1	168.5	190.2
3	19.9	<0.1	5.5	0.6	26.1	36.4
4 (center)	2.3	<0.1	0.8	<0.1	3.1	4.8

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values (in units of mrem/hr) are rounded to nearest tenths place.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

Table 5.1.19

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-37P WITH CE15X15 FUEL
LOADING PATTERNS (SEE TABLE 5.0.7)

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 HI-TRAC VW						
1	1594.4	21.7	698.5	53.6	2368.3	2719.1
2	4646.7	64.9	<0.1	115.7	4827.3	5901.4
3	46.4	6.4	477.6	8.0	538.3	1117.0
4	224.4	3.6	485.4	583.4	1296.7	1965.8
5	1287.2	5.3	2268.9	2180.3	5741.7	6026.3
ONE METER FROM THE HI-TRAC VW						
1	1063.8	7.5	90.7	18.4	1180.4	1430.2
2	2493.8	7.8	8.3	16.8	2526.7	3041.0
3	280.7	3.7	117.3	5.4	407.1	611.4
4	161.3	0.8	315.7	117.9	595.7	1025.3
5	836.8	1.3	991.0	434.9	2264.0	2401.0

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values (in units of mrem/hr) are rounded to the nearest tenths place.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 1 year cooling are used.

5.2.3.2 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. These devices are required to be stored in the locations as outlined in Subsection 2.1.

Subsection 5.4.4 discusses the effect on dose rate of the insertion of APSRs or CRAs into fuel assemblies.

5.2.4 Choice of Design Basis Assembly

The Westinghouse 17x17 and GE 10x10 assemblies were selected as design basis assemblies since they are widely used throughout the industry. Site specific shielding evaluations should verify that those assemblies and assembly parameters are appropriate for the site-specific analyses. Because of its large width, 16x16D (e.g., 16x16 Focus and 16x16 HTP fuel assemblies) was selected as design basis fuel assembly for MPC-32ML.

The active fuel length of Westinghouse 17x17 design basis fuel and Westinghouse 14x14 are identical, while the uranium mass (MTU) of Westinghouse 17x17 design basis fuel is about 15%

great~~higher~~. Therefore, Westinghouse 17x17 bounds 14x14 fuel and was utilized in shielding calculations in this Chapter.

As shown in Table 5.2.27 of reference [5.2.17], the gamma and neutron source terms as well as the decay heat for a Westinghouse 17x17 bound those for CE 15x15 fuel assemblies for a given burnup, enrichment, and cooling time and therefore Westinghouse 17x17 was utilized in source term~~hielding~~ calculations in this Chapter.

5.2.5 Decay Heat Loads and Allowable Burnup and Cooling Times

Subsection 2.1 describes the MPC maximum decay heat limits per assembly. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits.

5.2.6 Fuel Assembly Neutron Sources

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

During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. A detailed discussion about NSAs is provided in reference [5.2.17], where it is concluded that activation from NSAs ~~are~~is bounded by activation from BPRAs.

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Subsection 2.1. Further limitations allow for only one NSA to be stored in the MPC-37 (see Table 2.1.1a), or MPC-32ML (see Table 2.1.1b).

5.2.7 Design Basis Burnup and Cooling Times

For the fuel to be loaded into the HI-STORM FW system, the uniform and regionalized design basis loading curves (which specify burnup and cooling time combinations for each region of the cask) are provided in Tables 2.1.9, 2.1.10 and 2.1.1~~0~~2 using polynomial equation and corresponding polynomial coefficients.

In order to qualify the HI-STORM FW System with allowable burnup, cooling time combinations in Tables 2.1.9 and 2.1.10, the considered range of burnup, enrichment and cooling time combinations is selected as follows:

- 5 GWD/MTU burnup and burnups from 10 GWD/MTU to 70 GWD/MTU, in increments of 10 GWD/MTU for MPC-37 and MPC-89, and burnups from 15 GWD/MTU to 70 GWD/MTU, in increments of 5 GWD/MTU for MPC-32ML;

- The cooling time is calculated for each burnup using the equation and polynomial coefficients in Tables 2.1.9, 2.1.10 and 2.1.1~~0~~2. The determined cooling times are rounded down to the nearest available cooling time in the calculated source terms library, which provides a significant conservatism, especially, in the low cooling time area. For MPC-37 and MPC-89, the value of 1 year (minimum allowed cooling time) is used for all cooling times below 1 year. For MPC-32ML~~,~~ and MPC-44 the value of 3 year (minimum allowed cooling time) is used for all cooling times below 3 years.~~;~~ For MPC-37P the value of 1.6 year (minimum allowed cooling time) is used for all cooling times below 1.6 years;
- The appropriate burnup-specific lower bound enrichment is selected according to Table 5.2.17.

The final sets of the burnup, enrichment and cooling time combinations are provided in Tables 5.0.3 through 5.0.5.

5.2.8 Fuel Enrichment

As discussed in Subsection 5.2.2, enrichments have a significant impact on neutron dose rates, with lower enrichments resulting in higher dose rates at the same burnup. For assemblies with higher burnups (which result in high neutron source terms) and/or locations that are more neutron dominated, the enrichment would therefore be important in order to present dose rates in a conservative way. However, it would be impractical and excessively conservative to perform all calculations at bounding low enrichment, since low enrichments are generally only found in lower burned assemblies. Therefore, a conservatively low enrichment value is selected based on the burnup. Specifically, based on industry information on more than 130,000 PWR and 185,000 BWR assemblies, the fuel assemblies are distributed over different burnup range bins (0 5, 5-10 … 70-75 GWd/mtU). For instance, for a given burnup group of 5-10 GWd/mtU, the data array includes the enrichments for the fuel assemblies with the burnup from 5,000 MWd/mtU to 9,999 MWd/mtU. Then, in each burnup group, the array of enrichments is sorted from low to high, and the array index that precedes a fraction of 99% of the population is determined. The fuel enrichment under this array position represents the lower bound enrichment that conservatively bounds 99% of the fuel assembly population. The calculated and finally established lower bound enrichment values are summarized in Table 5.2.17.

Given that the considered baskets contain a relatively large number of available cells for fuel loading, selecting the minimum enrichment for all assemblies is considered reasonably conservative. The typical content of the basket would have most assemblies well above the lower bound enrichment assumed in the analyses, so even if a small number of assemblies would be below the assumed minimum, that would have a negligible effect or be essentially inconsequential for the dose rates around the cask. Furthermore, the site-specific shielding analysis shall consider actual or bounding fuel enrichment. Therefore, an explicit lower enrichment limit for the fuel assemblies is not considered necessary.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS CLAD FUEL

	PWR (MPC-37, MPC-37P and MPC-44)	**BWR (MPC-89)**
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	264	92
Rod pitch (in.)	0.496	0.51
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.374	0.404
Cladding thickness (in.)	0.0225	0.026
Pellet diameter (in.)	0.3232	0.345
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.522 (96% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Specific power (MW/MTU)	43.48	30
Weight of UO_2 (kg)††	532.150	213.531
Weight of U (kg)††	469.144	188.249
No. of Water Rods/ Guide Tubes	25	2
Water Rod/ Guide Tube O.D. (in.)	0.474	0.98
Water Rod/ Guide Tube Thickness (in.)	0.016	0.03

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR (MPC-37, MPC-37P and MPC-44)	**BWR (MPC-89)**
Lower End Fitting (kg)	5.9 (steel)	4.8 (steel)
Gas Plenum Springs (kg)	1.150 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.793 (inconel) 0.841 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	6.89 (steel) 0.96 (inconel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.6

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

Region	PWR (MPC-37, MPC-37P and MPC-44)	BWR (MPC-89)
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

Figure 5.3.14 shows a cross sectional view of the HI-TRAC VW Version V2 with the Neutron Shield Cylinder and MPC-89, as it was modeled in MCNP. Figure 5.3.15 shows a cross sectional view of the HI-TRAC VW Version V2 with the MPC-89, in which the MPC and annulus between the MPC and HI-TRAC inner cavity are filled with water, as it was modeled in MCNP.

Figure 5.3.16 shows a cross sectional view of the HI-STORM FW Version E with MPC-44, as it was modeled in MCNP. Figure 5.3.17 shows a cross sectional view of the HI-TRAC VW with the MPC-44, as it was modeled in MCNP. Figure 5.3.18 shows a cross sectional view of the HI-STORM FW Version E with MPC-37P, as it was modeled in MCNP. Figure 5.3.19 shows a cross sectional view of the HI-TRAC VW with the MPC-37P, as it was modeled in MCNP.

Calculations were performed for the HI-STORM 100 [5.2.17] to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it is acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR (in MPC-37) and BWR homogenized fuel assembly is equal to 17 times the pitch and 10 times the pitch, respectively. Homogenization results in a noticeable decrease in run time. The width of 16x16D fuel assembly in MCNP model of MPC-32ML is provided as a note under Table 5.3.1.

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

1. The fuel shims are not modeled because they are not needed on all fuel assembly types. However, most PWR fuel assemblies will have fuel shims. The fuel shim length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The MPC basket supports are not modeled. This is conservative since it removes material that would provide a small increase in shielding.

3. The MPC cavity height, MPC height and HI-STORM FW cavity height for HI-STORM FW with MPC-32ML are calculated using the length of fuel without non-fuel hardware and/or DFC, and data provided in Table 3.2.1.

Conservatively, the zircaloy flow channels are not included in the modeling of the BWR assemblies, unless explicitly mentioned.

Also, it should be noted that all dose calculations presented in this Chapter are performed with the HI-TRAC VW (standard) model unless otherwise noted. Site specific analysis of the HI-TRAC VW should consider the specific version of the HI-TRAC VW (for example, HI-TRAC VW (standard), HI-TRAC VW Version P, HI-TRAC VW Version V, HI-TRAC VW Version

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 (MPC-37, MPC-37P and MPC-44)					
Lower End Fitting	0.0	2.738	2.738	SS304	SS304
Space	2.738	3.738	1.0	zircaloy	void
Fuel	3.738	147.738	144.0	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	147.738	151.916	4.178	SS304 & inconel	SS304
Gas Plenum Spacer	151.916	156.095	4.179	SS304 & inconel	SS304
Upper End Fitting	156.095	159.765	3.670	SS304 & inconel	SS304
BWR (MPC-89)					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144.0	fuel & zircaloy	fuel & zircaloy
Space	151.385	157.385	6.0	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 fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
16x16D Fuel Region Mixture (MPC-32ML)	3.7565 (5.0 wt% U-235)	^{235}U	3.6111
		^{238}U	68.612
		O	9.709
		Zr	17.750
		Cr	0.262
		Fe	0.038
		Sn	0.018
Lower End Fitting (PWR MPC-37, MPC-37P and MPC-44)	1.849	SS304	100
Gas Plenum Springs (PWR MPC-37, MPC-37P and MPC-44)	0.23626	SS304	100
Gas Plenum Spacer (PWR MPC-37, MPC-37P and MPC-44)	0.33559	SS304	100
Upper End Fitting (PWR MPC-37, MPC-37P and MPC-44)	1.8359	SS304	100
Lower End Fitting (BWR)	1.5249	SS304	100
Gas Plenum Springs (BWR)	0.27223	SS304	100
Expansion Springs (BWR)	0.69514	SS304	100

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Upper End Fitting (BWR)	1.4049	SS304	100
Handle (BWR)	0.26391	SS304	100
Lower End Fitting (MPC-32ML)	0.6022	SS304	100
Gas Plenum Springs (MPC-32ML)	0.159	SS304	100
Gas Plenum Spacer (MPC-32ML)	0.159	SS304	100
Upper End Fitting (MPC-32ML)	1.0032	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.919 (water jacket)	H	11.2
	0.958 (inside MPC)	O	88.8
PWR Fuel Region Mixture (MPC-44)	3.769 (5.0 wt% U-235)	235U	3.683
		238U	69.973
		O	9.902
		Zr	16.153
		Cr	0.016
		Fe	0.035
		Sn	0.238

Proprietary Information Withheld in Accordance with 10 CFR 2.390

Figure 5.3.16
HI-STORM FW VERSION E OVERPACK WITH MPC-44 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

Proprietary Information Withheld in Accordance with 10 CFR 2.390

Figure 5.3.17
HI-TRAC VW OVERPACK WITH MPC-44 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

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Figure 5.3.18
HI-STORM FW VERSION E OVERPACK WITH MPC-37P CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

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Figure 5.3.19
HI-TRAC VW OVERPACK WITH MPC-37P CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

[5.2.8] O. W. Hermann, et al., "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

[5.2.9] M. D. DeHart and O. W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Oak Ridge National Laboratory, September 1996.

[5.2.10] O. W. Hermann and M. D. DeHart, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel," ORNL/TM-13315, Oak Ridge National Laboratory, September 1998.

[5.2.11] "Summary Report of SNF Isotopic Comparisons for the Disposal Criticality Analysis Methodology," B00000000-01717-5705-00077 REV 00, CRWMS M&O, September 1997.

[5.2.12] "Isotopic and Criticality Validation of PWR Actinide-Only Burnup Credit," DOE/RW-0497, U.S. Department of Energy, May 1997.

[5.2.13] B. D. Murphy, "Prediction of the Isotopic Composition of UO_2 Fuel from a BWR: Analysis of the DU1 Sample from the Dodewaard Reactor," ORNL/TM-13687, Oak Ridge National Laboratory, October 1998.

[5.2.14] O. W. Hermann, et al., "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data," NUREG/CR-5625, ORNL-6698, Oak Ridge National Laboratory, September 1994.

[5.2.15] C. E. Sanders, I. C. Gauld, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor," NUREG/CR-6798, ORNL/TM-2001/259, Oak Ridge National Laboratory, January 2003.

[5.2.16] Not Used.

[5.2.17] HI-2002444, Proposed Rev. 21~~17A~~, "Final Safety Analysis Report for the HI-STORM 100 Cask System", USNRC Docket 72-1014, Submittal Letter 5014917~~878~~, submitted on March~~September~~ 9~~16~~, 2021~~19~~.

[5.2.18] Safety Analysis Report on the HI-STAR 190 Package, Holtec International Report HI 2146214, Revision 3, USNRC Docket No 71-9373, Washington, DC.

[5.4.1] "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors", ANSI/ANS-6.1.1-1977.

TABLE 6.1.1 (c)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN MPC-44 (HI-TRAC VW)

Fuel Assembly Class	**4.0 wt% ^{235}U Maximum Enrichment†**		**5.0 wt% ^{235}U Maximum Enrichment†**	
	Minimum Soluble Boron Concentration (ppm)	**Maximum k_{eff}**	**Minimum Soluble Boron Concentration (ppm)**	**Maximum k_{eff}**
14x14A	1400	0.8956	1900	0.9147
14x14B	1400	0.9225	1900	0.9436

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-89 (HI-TRAC VW)

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
7x7B	4.8	0.9317
7x7C	4.8	0.9318
8x8B	4.8	0.9369
8x8C	4.8	0.9399
8x8D	4.8	0.9380
8x8E	4.8	0.9281
8x8F	4.5	0.9328
8x8G	4.8	0.9301
9x9A	4.8	0.9421
9x9B	4.8	0.9410
9x9C	4.8	0.9338
9x9D	4.8	0.9342
9x9E/F	4.5	0.9346
9x9G	4.8	0.9307
10x10A	4.8	0.9435
10x10B	4.8	0.9417
10x10C	4.8	0.9389
10x10F	4.7	0.9440
10x10G	4.6	0.9466
10x10I	4.8	0.9422
10x10J	4.8	0.9477
11x11A	4.8	0.9457

TABLE 6.1.4(c)

BOUNDING MAXIMUM k_{eff} VALUES FOR MPC-44
WITH UP TO 12 DFC/DFIs*

Fuel Assembly Class of Undamaged Fuel	4.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris †			5.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris †		
	Minimum Soluble Boron Concentration (ppm)	DFC	DFI	Minimum Soluble Boron Concentration (ppm)	DFC	DFI
14x14A and 14x14B	1500	0.9160	0.9147	2000	0.9384	0.9364

* The permissible location of DFC/DFIs is provided in Figure 2.1.1d. DFIs are restricted to damaged fuel only.

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-89
WITH UP TO 16 DFC/DFIs*

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	DFC	DFI
All BWR Classes except 8x8F, 9x9E/F, 10x10F, ~~and~~ 10x10G, 10x10I, 10x10J and 11x11A	4.8	0.9464	0.9421
8x8F, 9x9E/F and 10x10G	4.0	0.9299	0.9074
10x10F	4.6	0.9428	0.9372
10x10I, 10x10J and 11x11A	4.7	0.9451~~0.9432~~	~~0.9393~~0.9400

* The permissible location of DFC/DFIs is provided in Figure 2.1.2. DFIs are restricted to damaged fuel only.

TABLE 6.1.8

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-89
WITH UP TO 12 DFC/DFIs AND 4 EMPTY CELLS*

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	DFC	DFI
All BWR Classes except 8x8F, 9x9E/F, 10x10F, 10x10G, 10x10I, 10x10J and 11x11A	4.8	0.9305	0.9269
8x8F, 9x9E/F and 10x10G	4.0	0.8977	0.8912
10x10F	4.6	0.9271	0.9217
10x10I, 10x10J~~-~~and 11x11A	4.7	~~0.9297~~0.9319	~~0.9245~~0.9248

* The permissible location of DFC/DFIs is provided in Figure 1.2.6b. DFIs are restricted to damaged fuel only.

Since all assemblies have the same principal design, i.e. consist of bundles of cladded fuel rods, most of them with embedded guide/instrument tubes or water rods or channels, the above conclusions apply to all of them, and the bounding dimensions are therefore also common to all fuel assemblies analyzed here. Nevertheless, to clearly demonstrate that the main assumption is true, i.e. that all assemblies are undermoderated, a study was performed for all assembly types where the pellet-to-clad gap is empty instead of being flooded (a conservative assumption for the design basis calculations, see Section 6.4.2.3) The results are listed in Table 6.2.3, in comparison with the results of the reference cases with the flooded gap from Section 6.1 for those assembly types. In all cases, the reactivity is reduced compared to the reference case. This verifies that all assembly types considered here are in fact undermoderated, and therefore validates the main assumption stated above. All assembly types are therefore behaving in a similar fashion, and the bounding dimensions are therefore applicable to all assembly types. This discussion and the corresponding conclusions not only affect fuel behavior, but also other moderation effects, and is therefore further referenced in Section 6.3.1 and 6.4.2

As a result, the authorized contents in Subsection 2.1 are defined in terms of those bounding assembly parameters for each class.

Nevertheless, to further demonstrate that the aforementioned characteristics are in fact bounding for the HI-STORM FW, parametric studies were performed on reference PWR and BWR assemblies, namely PWR assembly class 17x17B and BWR assembly class 10x10A. The results of these studies are shown in Table 6.2.1 and 6.2.2, and verify the bounding parameters listed above. Note that in the studies presented in Tables 6.2.1 and 6.2.2, the fuel pellet diameter and cladding inner diameter are changed together. This is to keep the cladding-to-pellet gap, which is conservatively flooded with pure water in all cases (see Section 6.4.2.3), at a constant thickness, to ensure the studies evaluate the fuel parameters rather than the moderation conditions, as discussed above.

In addition to those dimensions, additional fuel assembly characteristics important to criticality control are the location of guide tubes, water rods, part length rods, and rods with differing dimensions (classes 9x9E/F only). These are identified in the assembly cross sections provided in Appendix 6.B, Section B.4.

In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel, and Integral Fuel Burnable Absorbers (IFBA) used in PWR fuel was conservatively neglected.

Some assembly classes contain partial length rods. There are differences in location of those partial length rods within the assembly that influence how those rods affect reactivity: Assembly classes 9x9A, 10x10A, 10x10B and 10x10F have partial length rods that are completely surrounded by full length rods, whereas assembly classes 10x10G and 10x10J have~~s~~ those partial length rods on the periphery of the assembly or facing the water gap, where they directly only face two full length rods (see Appendix 6.B, Section B.4). Assembly classes 10x10I and 11x11A have both types of the partial length rod locations. To determine a bounding configuration for those assembly classes where partial length rods are completely surrounded by

full length rods, calculations are listed in Table 6.2.2 for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. The results show that the configurations with only the full length rods present, i.e. where the partial length rods are assumed completely absent from the assembly, is bounding. This is an expected outcome, since LWR assemblies are typically undermoderated, therefore reducing the fuel-to-water-ratio within the rod array tends to increase reactivity. Consequently, all assembly classes that contain partial length rods surrounded by full-length rods are analyzed with the partial length rods absent. For assembly class 10x10G, calculations with different assumptions for the length of the part-length rods are presented in Table 6.2.7, and show that reducing the length of the part length rods reduces reactivity. This means that the reduction in the fuel amount is more dominating than the change in moderation for this configuration. For this class, all rods therefore are assumed full length. For assembly classes 10x10I, 10x10J and 11x11A, where it is not clear which type of the partial length rod location is dominating, calculations for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods are presented in Table 6.2.8, and show that the configurations with only the full length rods present, i.e. where the partial length rods are assumed completely absent from the assembly, is bounding for assembly classes 10x10I and 11x11A, and the configuration with the actual (real) assembly configuration is bounding for assembly class 10x10J. Additional studies are further performed for assembly class 10x10J with variations in the lengths of the short and long partial length rods, results are presented in Table 6.2.9 and show that the configuration with the short partial length rods (facing the water rod) completely removed but the long partial length rods (on the periphery of the assembly) extended to full length is bounding. Therefore, this condition is used in the design basis calculations of the assembly class 10x10J. Note that in neither of the cases is the configuration with the actual part length rods bounding. The specification of the authorized contents has therefore no minimum requirement for the active fuel length of the partial length rods.

BWR assemblies are specified in Table 2.1.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in the HI-STORM FSAR ([6.0.1], Chapter 6, Appendix 6.B) show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum k_{eff}. To verify that this is applicable to the HI-STORM FW, those calculations were re-performed in the MPC-89. The results are presented in Table 6.2.4, and show that, as expected, the planar average enrichments bound or are statistically equivalent to the distributed enrichment in the HI-STORM FW as they do in the HI-STORM 100. To confirm that this is also true for the higher enrichments analyzed here, additional calculations were performed and are presented in Table 6.2.2 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt% ^{235}U is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt% ^{235}U is close to the maximum rod enrichment of 5.0 wt% ^{235}U, the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt% ^{235}U, while the balance of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are

performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results are also included in table 6.2.2 and show that in all cases, the maximum k_{eff} calculated for the distributed enrichments are statistically equivalent to or below those for the uniform enrichments. Therefore, modeling BWR assemblies with distributed enrichments using a uniform enrichment equal to the planar-average value is acceptable and conservative. The assumed enrichment distributions analyzed are shown in Appendix 6.B.

Note that for some BWR fuel assembly classes, 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.

Two BWR classes (8x8B and 8x8D) are specified with slight variation in the number of fuel and/or water rods (see Section 6.B.4). The results listed in Section 6.1 utilize the minimum number of fuel rods, i.e. maximizing the water-to-fuel ratio. To show that this is appropriate and bounding, calculations were also performed with the alternative configurations, and are presented in Table 6.2.5. The results show that the reference conditions used for the calculations documented in Section 6.1 are in fact bounding.

For BWR assembly class 9x9E/F, two patterns of water rods were analyzed (see Section 6.B.4). The comparison is also presented in Table 6.2.5 and shows that the condition with the larger water rod spacing is bounding.

For BWR assembly class 10x10J, a water rod may be with a variable diameter that displaces 4 or 12 fuel rods. But in all the calculations of this analysis, a water rod segment that displaces 4 fuel rods is assumed along the entire active fuel length. This is conservative since smaller water rod contains less material thus displaces less water, and the amount of fissile material may be potentially larger.

For PWR assembly class 15x15I (see Section 6.B.4), calculations with and without guide rods were performed. The comparison is also presented in Table 6.2.5. The case without the guide rods is used as the design basis case for this assembly type, therefore, no specific restrictions on the location and number of guide rods exists. The 15x15I fuel assembly class also includes versions with reduced number of fuel rods in specific locations, specifically; two versions with 212 fuel rods and one version with 208 fuel rods (see Section 6.B.4 for the specific fuel cross-sections). The missing fuel rods can be replaced with water or guide tubes. The comparison for the reduced number of fuel rods in 15x15I fuel class is shown in Table 6.2.5.

Typically, PWR fuel assemblies are designed with solid fuel pellets throughout the entire active fuel length. However, some PWR assemblies contain annular fuel pellets in the top and bottom 6

TABLE 6.2.3

EFFECT OF THE FLOODING OF THE PELLET-TO-CLAD GAP

Fuel Assembly Class	**Maximum k_{eff} at 5.0 wt% ^{235}U Maximum Enrichment**		
	Flooded Pellet-to-Clad Gap	**Empty Pellet-to-Clad Gap**	**Difference**
MPC-37			
14x14A	0.8983	0.8962	-0.0021
14x14B	0.9282	0.9235	-0.0047
14x14C	0.9277	0.9237	-0.0038
15x15B	0.9311	0.9284	-0.0027
15x15C	0.9188	0.9164	-0.0024
15x15D	0.9421	0.9386	-0.0035
15x15E	0.9410	0.9371	-0.0039
15x15F	0.9455	0.9408	-0.0047
15x15H	0.9325	0.9300	-0.0025
15x15I	0.9357	0.9305	-0.0052
16x16A	0.9366	0.9284	-0.0082
16x16A[DFC]	0.9400	0.9340	-0.0060
16x16B	0.9334	0.9297	-0.0037
16x16C	0.9187	0.9144	-0.0043
~~16x16D~~	~~0.9427~~	~~0.9361~~	~~-0.0066~~
16x16E	0.9303	0.9223	-0.0080
17x17A	0.9194	0.9160	-0.0034
17x17B	0.9380	0.9335	-0.0045
17x17C	0.9424	0.9375	-0.0049
17x17D	0.9384	0.9323	-0.0061
17x17E	0.9392	0.9346	-0.0046

TABLE 6.2.3 (continued)

EFFECT OF THE FLOODING OF THE PELLET-TO-CLAD GAP

Fuel Assembly Class	Maximum k_{eff}		
	Flooded Pellet-to-Clad Gap	Empty Pellet-to-Clad Gap	Difference
MPC-89			
7x7B	0.9317	0.9261	-0.0056
7x7C	0.9318	0.9263	-0.0055
8x8B	0.9369	0.9318	-0.0051
8x8C	0.9399	0.9331	-0.0068
8x8D	0.9380	0.9334	-0.0046
8x8E	0.9281	0.9230	-0.0051
8x8F	0.9328	0.9275	-0.0053
8x8G	0.9301	0.9240	-0.0061
9x9A	0.9421	0.9370	-0.0051
9x9B	0.9410	0.9292	-0.0118
9x9C	0.9338	0.9290	-0.0048
9x9D	0.9342	0.9294	-0.0048
9x9E/F	0.9346	0.9261	-0.0085
9x9G	0.9307	0.9250	-0.0057
10x10A	0.9435	0.9391	-0.0044
10x10B	0.9417	0.9317	-0.0100
10x10C	0.9389	0.9333	-0.0056
10x10F	0.9440	0.9395	-0.0045
10x10G	0.9466	0.9408	-0.0058
10x10I	0.9422	0.9383	-0.0039
10x10J	0.9477	0.9419	-0.0058
11x11A	0.9457	0.9421	-0.0036

TABLE 6.2.3 (continued)

EFFECT OF THE FLOODING OF THE PELLET-TO-CLAD GAP

Fuel Assembly Class	**Maximum k_{eff}**		
	Flooded Pellet-to-Clad Gap	**Empty Pellet-to-Clad Gap**	**Difference**
MPC-32ML			
16x16D	0.9427	0.9361	-0.0066
MPC-44			
14x14A	0.9147	0.9117	-0.0030
14x14B	0.9436	0.9400	-0.0036

Table 6.2.8

EFFECT OF PARTIAL LENGTH RODS FOR ASSEMBLY CLASSES 10x10I, 10x10J AND 11x11A

Parameter Variation	reactivity effect	Maximum k_{eff}	standard deviation
Assembly Class 10x10I			
full-length rods only	Reference	0.9422	0.0006
full-length and part-length rods (real assembly)	-0.0041	0.9381	0.0006
part-length rods extended to full-length	-0.0117	0.9305	0.0005
Assembly Class 11x11A			
full-length rods only	Reference	0.9457	0.0006
full-length and part-length rods (real assembly)	-0.0040	0.9417	0.0005
part-length rods extended to full-length	-0.0062	0.9395	0.0006
Assembly Class 10x10J			
full-length rods only	-0.0151	0.9298	0.0003
full-length and part-length rods (real assembly)	Reference	0.9449	0.0003
part-length rods extended to full-length	-0.0015	0.9434	0.0003

Table 6.2.9

ADDITIONAL STUDY OF PARTIAL LENGTH RODS EFFECT FOR ASSEMBLY CLASS 10x10J

Length of Long Partial Length Rod	**Length of Short Partial Length Rod**	**Reactivity Effect**	**Maximum k_{eff}**	**Standard Deviation**
0"	0"	-0.0179	0.9298	0.0003
0"	Real Assembly	-0.0190	0.9287	0.0003
0"	Extended to Full-Length	-0.0203	0.9274	0.0003
Real Assembly	0"	-0.0015	0.9462	0.0003
Real Assembly	Real Assembly	-0.0028	0.9449	0.0003
Real Assembly	Extended to Full-Length	-0.0059	0.9418	0.0003
Extended to Full-Length	0"	Reference	0.9477	0.0003
Extended to Full-Length	Real Assembly	-0.0015	0.9462	0.0003
Extended to Full-Length	Extended to Full-Length	-0.0043	0.9434	0.0003

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1 through 6.3.9~~7~~ show representative cross sections of the criticality models for all considered baskets. Figures 6.3.1, 6.3.2, 6.3.6 and 6.3.8~~6~~ show single cell from each basket. Figures 6.3.3, 6.3.4, ~~and~~ 6.3.7 and 6.3.9 show the entire MPC-37, MPC-89, ~~and~~ MPC-32ML and MPC-44 basket, respectively. Figure 6.3.5 shows a sketch of the calculational model in the axial direction.

Full three-dimensional calculational models were used for all calculations. The calculational models explicitly define the fuel rods and cladding, the guide tubes, water rods and the channel (for the BWR assembly), the neutron absorber walls of the basket cells, and the surrounding MPC shell and overpack. For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel rod pellet-to-clad gaps, since this represents the bounding condition as demonstrated in Section 6.4.2.3. Appendix 6.B provides sample input files for typical MPC basket designs

Note that the water thickness above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR and PWR assembly type, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion in Section 6.2.1 are listed below. Note that this approach is consistent with that used for the HI-STORM 100. ~~The MPC-32ML basket design is very similar to the MPC-37 basket; the only major difference is the increased cell ID and, consequently, the number of storage locations is reduced. Therefore, all studies performed for MPC-37 are directly applicable to MPC-32ML, since the same behavior (reactivity effect) is expected. However, MPC-32ML basket specific studies are performed and discussed below.~~

PWR fuel assembly class 15x15I which is authorized for the MPC-37 is also qualified for a slightly modified iteration of the MPC-37, designated as MPC-37P, wherein the box ID is slightly decreased, and the box wall thickness is increased, so that the basket cell pitch is essentially maintained. Table 2.1.3 provides the acceptable fuel characteristics for the assembly class 15x15I authorized for storage in the MPC-37P. From a criticality safety perspective, the MPC-37 basket designs bound the MPC-37P basket designs. As presented in Table 6.3.2, for the MPC-37P, though the box ID is decreased in comparison to the MPC-37 which reduces the amount of moderator (water) between the basket walls and fuel assembly thus reduce efficiency of the fixed neutron absorbers and the amount of the soluble boron, such effect is compensated by the substantially larger amount of neutron absorbers with the increased box wall thickness. Overall, the reactivity of the MPC-37P is bounded by the MPC-37, as confirmed by results

shown in Table 6.3.2. Therefore, the results and conclusions of the design basis criticality safety calculations and studies for the MPC-37 presented in this chapter apply to the MPC-37P. No additional analyses are required for the MPC-37P in this chapter.

Basket Manufacturing Tolerance: The two aspects of the basket tolerance that are evaluated are the cell wall thickness and the cell ID. The reduced cell wall thickness results in a reduced amount of poison (since the material composition of the wall is fixed), and therefore in an increase in reactivity. The reduced cell ID reduces amount of water between the fuel the poison, and therefore the effectiveness of the poison material. Both effects are simply a function of the geometry, and are independent of the fuel type.

Panel Gaps: Similar to the basket manufacturing tolerance for the cell wall thickness, this tolerance has a small effect on the overall poison amount of the basket, which would affect the reactivity of the system independent of the fuel type.

Eccentric positioning (see Section 6.3.3): When a fuel assembly is located in the center of a basket cell, it is surrounded by equal amounts of water on all sides, and hence the thermalization of the neutrons that occur between the assembly and the poison in the cell wall, and hence the effectiveness of the poison, is also equal on all sides. For an eccentric positioning, the effectiveness of the poison is now reduced on those sides where the assembly is located close to the cell walls, and increased on the opposite sides. This creates a compensatory situation for a single cell, where the net effect is not immediately clear. However, for the entire basket, and for the condition where all assemblies are located closest to the center of the basket, the assemblies at the center of the basket are now located close to each other, separated by poison plates with a reduced effectiveness since they are not surrounded by water on any side. This now becomes the dominating condition is terms of reactivity increase. This effect is also applicable to all assembly types, since those assemblies are all located close to the center of the basket, i.e. the eccentric position with all assemblies moved towards the center will be bounding regardless of the assembly type.

Wall thicknesses of DFCs: DFCs are used for damaged fuel and fuel debris in selected locations of the basket, but are also permitted to be used for intact fuel of the array/type 16x16A. Generally, DFCs are thin-walled containers, in order to minimize the additional weight to be supported by the basket. For damaged fuel and fuel debris calculations, a wall thickness of 0.025” is used, and studies with larger wall thicknesses show an insignificant effect. However, when DFCs are also used for intact assemblies of the 16x16A array/type, it is shown that there is a noticeable effect when a thicker wall is modeled. Consequently, for DFCs for intact 16x16A fuel, a conservative DFC wall thickness of 0.075” is used to provide manufacturing flexibility. Note that this already a rather thick wall, the same as typically used for storage racks in spent fuel pools, and would therefore present a practical upper limit for any DFC design.

The basket geometry can vary due to manufacturing tolerances and due to potential deflections of basket walls as the result of accident conditions. The basket tolerances are defined on the drawings in Chapter 1. The structural acceptance criteria for the basket during accident

2000 ppm soluble boron in the water in the MPC-37, ~~and~~ for the assembly class 16x16D with 2000 ppm soluble boron in the water in the MPC-32ML, and for the assembly class 14x14B with 1900 ppm soluble boron in the water in the MPC-44. These are the same assemblies and conditions used for the fuel dimension studies in Section 6.2, and shown there to be representative of all assemblies qualified for those baskets. The results are presented in Table 6.3.1, and show that the minimum water temperature (corresponding to a maximum water density) are bounding. This condition is therefore used in all further calculations. This is expected since an increased temperature results in a reduced water density, a condition that is shown in Section 6.4 to result in reduced reactivities.

Calculations documented in Chapter 3 show that the baskets stay within the applicable structural limits during all normal and accident conditions. Furthermore, the neutron poison material is an integral and non-removable part of the basket material, and its presence is therefore not affected by the accident conditions. Except for the potential deflection of the basket walls that is already considered in the criticality models, damage to the cask under accident conditions is limited to possible loss of the water in the water jacket of HI-TRAC VW. However, this condition is already considered in the calculational models. Other parameters important to criticality safety are fuel type and enrichment, which are not affected by the hypothetical accident conditions. The calculational models of the cask and basket for the accident conditions are therefore identical to the models for normal conditions, and no separate models need to be developed for accident conditions.

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM FW system are listed in Table 6.3.4. The cross section set for each nuclide is listed in Table 6.3.8, and is consistent with the cross section sets used in the benchmarking calculations documented in Appendix A. Note that these are the default cross sections chosen by the code.

The HI-STORM FW system is designed such that the fixed neutron absorber will remain effective for a storage period greater than 60 years, and there are no credible means to lose it.

The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Subsection 10.1.6.3, 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 evaluation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 60 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 less than 10^{-7} in 60 years. Thus, the reduction in ^{10}B concentration in the fixed neutron absorber by neutron absorption is negligible. 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.

TABLE 6.3.1

CASMO-4 CALCULATIONS FOR EFFECT OF TEMPERATURE

Change in Nominal Parameter	**Δk Maximum Tolerance**				**Action/Modeling Assumption**
	MPC-37, 17x17B, 5.0 wt%, Borated Water with 2000 ppm Soluble Boron	**MPC-89, 10x10A, 4.8 wt%, Fresh Water**	**MPC-32ML, 16x16D, 5.0 wt%, Borated Water with 2000 ppm Soluble Boron**	**MPC-44, 14x14B, 5.0 wt%, Borated Water with 1900 ppm Soluble Boron**	
Increase in Temperature 20°C 40°C 70°C 100°C	Ref. -0.0008 -0.0023 -0.0042	Ref. -0.0035 -0.0100 -0.0180	Ref. -0.0004 -0.0012 -0.0023	Ref. -0.0007 -0.0021 -0.0039	Assume 20°C
10% Void in Moderator 20°C with no void 20°C 100°C	Ref. -0.0036 -0.0096	Ref. -0.0282 -0.0463	Ref. -0.0010 -0.0055	Ref. -0.0028 -0.0087	Assume no void

TABLE 6.3.2

EVALUATION OF BASKET MANUFACTURING TOLERANCES

Box I.D.	**Box Wall Thickness**	**Maximum k_{eff}**
MPC-37 (17x17B, 5.0% Enrichment)		
nominal (8.94")	nominal (0.59")	0.9332
nominal (8.94")	minimum (0.57")	0.9346
increased (8.96")	minimum (0.57")	0.9350
minimum (8.92")	minimum (0.57")	0.9352
minimum, including deformation (8.875")	minimum (0.57")	0.9374
MPC-37P (17x17B †, 5.0% Enrichment)		
minimum, including deformation (8.656")	minimum (0.77")	0.9300
MPC-89 (10x10A 4.8% Enrichment)		
nominal (6.01")	nominal (0.40")	0.9365
nominal (6.01")	minimum (0.38")	0.9403
increased (6.03")	minimum (0.38")	0.9396
minimum (5.99")	minimum (0.38")	0.9417
minimum, including deformation (5.96")	minimum (0.38")	0.9428
MPC-32ML (16x16D, 5.0% Enrichment)		
nominal (9.57")	nominal (0.59")	0.9366
nominal (9.57")	minimum (0.57")	0.9385
increased (9.61")	minimum (0.57")	0.9367
minimum (9.53")	minimum (0.57")	0.9410
minimum, including deformation (9.482")	minimum (0.57")	0.9428
MPC-44 (14x14B, 5.0% Enrichment)		
nominal (8.10")	nominal (0.51")	0.9358
nominal (8.10")	minimum (0.49")	0.9389
increased (8.20")	minimum (0.49")	0.9348
minimum (8.00")	minimum (0.49")	0.9425
minimum, including deformation (7.9595")	minimum (0.49")	0.9436

† Assembly class 17x17B is used in the analysis (instead of 15x15I that to be authorized to be loaded) for the MPC-37P, this is acceptable because in this table, the result for the MPC-37P is only compared with that for the MPC-37 also analyzed with 17x17B to show the reactivity of the MPC-37P is bounded by the MPC-37.

TABLE 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Box I.D.	Box Wall Thickness
MPC-37	minimum, including deformation(8.875")	minimum (0.57”)
MPC-89	minimum, including deformation (5.96")	minimum (0.38”)
MPC-32ML	minimum, including deformation(9.482")	minimum (0.57”)
MPC-44	minimum, including deformation(7.9595")	minimum (0.49”)

TABLE 6.3.5 (Continued)

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT (FUEL ASSEMBLIES AND DFC/DFIs) 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-44, Undamaged Fuel	0.9405	0.9436	0.0031	0.9351	-0.0054
MPC-44, Undamaged Fuel and Damaged Fuel/Fuel Debris (12 DFCs)	0.9362	0.9384	0.0022	0.9317	-0.0045
MPC-37, Undamaged Fuel and Damaged Fuel (12 DFIs)	0.9279	0.9110	-0.0169	0.9105	-0.0174
MPC-89, Undamaged Fuel and Damaged Fuel (16 DFIs)	0.9421	0.9388	-0.0033	0.9236	-0.0185
MPC-32ML, Undamaged Fuel and Damaged Fuel (8 DFIs)	0.9381	0.9281	-0.0100	0.9179	-0.0202
MPC-44, Undamaged Fuel and Damaged Fuel (12 DFIs)	0.9364	0.9271	-0.0093	0.9291	-0.0073

TABLE 6.3.6

REACTIVITY EFFECTS GAPS IN BASKET CELL PLATES

Cases	Maximum k_{eff}	
	Gaps in Metamic-HT	
	None	0.06" every 10"
MPC-37 (17x17B, 5.0% ENRICHMENT)	0.9380	0.9382
MPC-89 (10x10A, 4.8% ENRICHMENT)	0.9435	0.9439
MPC-32ML (16x16D, 5.0% ENRICHMENT)	0.9427	0.9426
MPC-44 (14x14B, 5.0% ENRICHMENT)	0.9436	0.9431

~~Gaps in Metamic-HT~~	~~MPC-37 (17x17B, 5.0% ENRICHMENT)~~	~~MPC-89 (10x10A, 4.8% ENRICHMENT)~~	~~MPC-32ML (16x16D, 5.0% ENRICHMENT~~	
~~None~~	~~0.9380~~	~~0.9435~~	~~0.9427~~	
~~0.06" every 10"~~	~~0.9382~~	~~0.9439~~	~~0.9426~~	

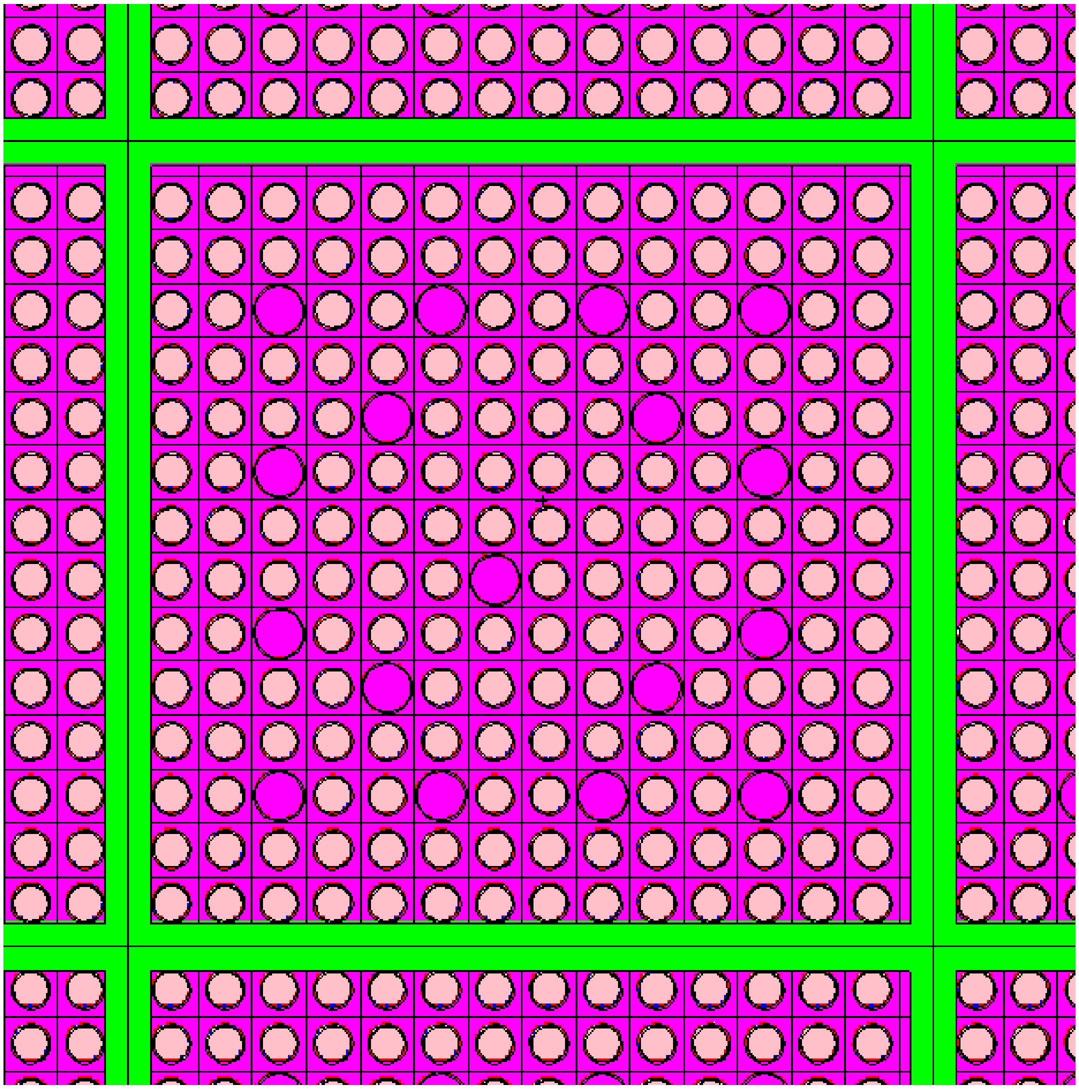

Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.8: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in MPC-44

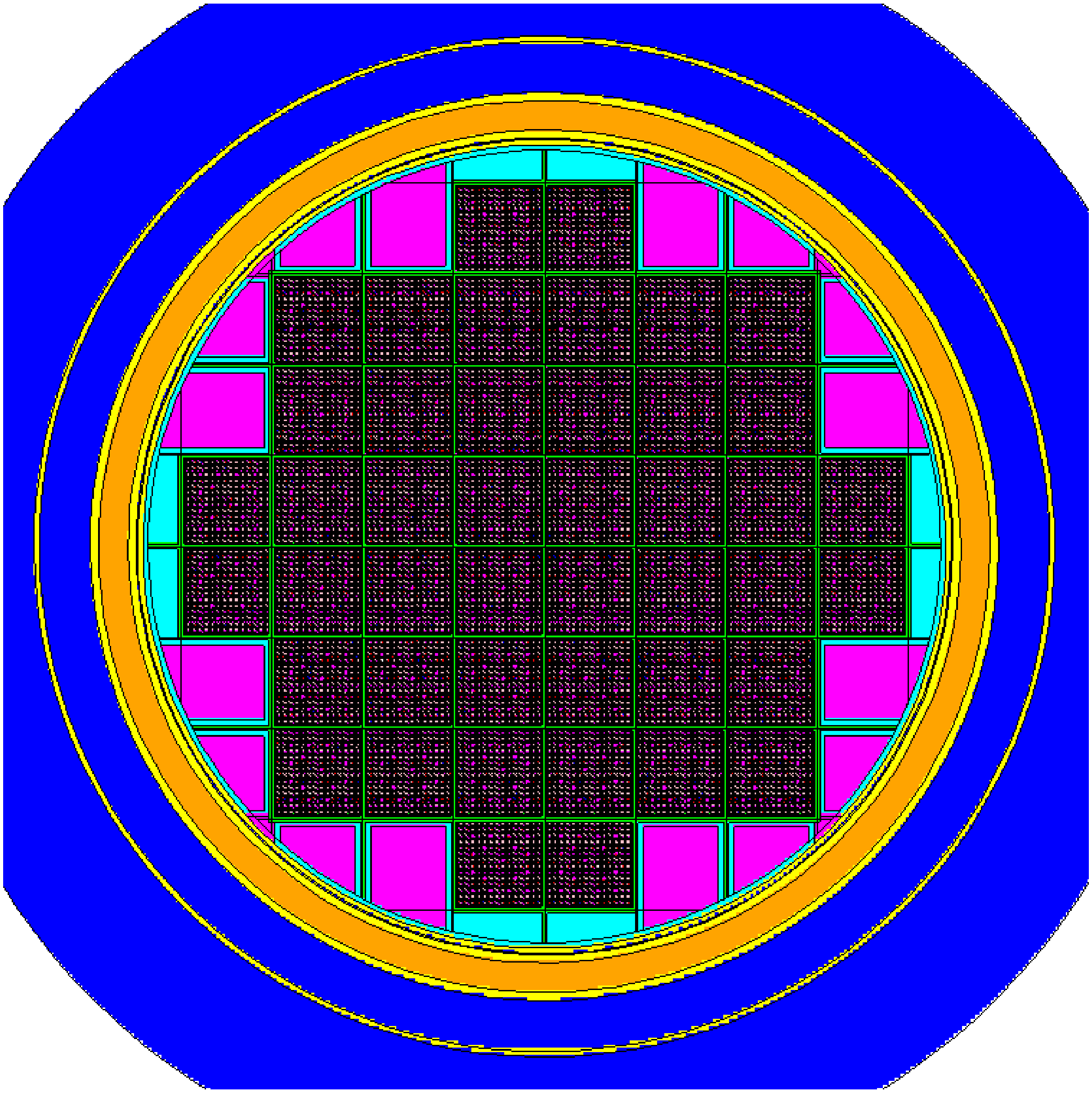

Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.9: Calculational Model (planar cross-section) of MPC-44

To identify the configuration or configurations leading to the highest reactivity, 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 subsections.

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.

Note that the modeling approach for damaged fuel and fuel debris is identical to that used in the HI-STORM 100 and HI-STAR 100.

Bounding Undamaged Assemblies

The undamaged assemblies assumed in the basket in those cells not filled with DFCs or DFIs are those that show the highest reactivity for each group of assemblies, namely

- 9x9E for BWR 9x9E/F, 8x8F and 10x10G assemblies
- 10x10F for BWR 10x10F assemblies
- 10x10J~~11x11A~~ for BWR 10x10I, 10x10J and 11x11A assemblies;
- 10x10A for all other BWR assemblies;
- 16x16A for all PWR assemblies with 14x14 and 16x16 arrays;
- 15x15F for all PWR assemblies with 15x15 and 17x17 arrays; ~~and~~
- 16x16D for all PWR assemblies qualified for MPC-32ML; and~~.~~
- 14x14B for all PWR assemblies qualified for MPC-44.

Since the damaged fuel modeling approach results in higher reactivities, requirements of soluble boron for PWR fuel and maximum enrichment for BWR fuel are different from those for undamaged fuel only. Those limits are listed in Table 6.1.4 (PWR) and Table 6.1.5 (BWR) in Section 6.1. Also, those limits are applicable to the basket loading configurations, considered in Tables 6.1.7 (PWR) and Tables 6.1.8 (BWR) in Section 6.1. Note that for the calculational cases for damaged and undamaged fuel in the MPC-89, the same enrichment is used for the damage and undamaged assemblies.

Note that for the first group of BWR assemblies listed above (9x9E/F, 8x8F and 10x10G), calculations were performed for both 9x9E and 10x10G as undamaged assemblies, and assembly class 9x9E showed the higher reactivity, and is therefore used in the design basis analyses. This may seem contradictory to the results for undamaged assemblies listed in Table 6.1.2, where the 10x10G shows a higher reactivity. However, the cases in Table 6.1.2 are not at the same enrichment between those assemblies.

All calculations with damaged and undamaged fuel are performed for an active length of 150 inches. There are two assembly classes (17x17D and 17x17E) that have a larger active length for the undamaged fuel. However, the calculations for undamaged fuel presented in Table 6.1.1 show that the reactivity of those undamaged assemblies is at least 0.0050 delta-k lower than that

As an example of the damaged fuel model used in the analyses, Figure 6.4.1 shows the basket cell of an MPC-37 with a DFC or DFI containing a 14x14 array of bare fuel rods.

Principal results are listed in Tables 6.4.6, 6.4.7, ~~and~~ 6.4.11 and 6.4.12 for the MPC-37, MPC-89, ~~and~~ MPC-32ML and MPC-44, respectively. In all cases, the maximum k_{eff} is below the regulatory limit of 0.95.

For the HI-STORM 100, additional studies for damaged fuel assemblies were performed to further show that the above approach using arrays of bare fuel rods are bounding. The studies considered conditions including

- Fuel assemblies that are undamaged except for various numbers of missing rods
- Variations in the diameter of the bare fuel rods in the arrays
- Consolidated fuel assemblies with cladded rods
- Enrichment variations in BWR assemblies

Results of those studies were shown in the HI-STORM 100 FSAR, Table 6.4.8 and 6.4.9 and Figure 6.4.13 and 6.4.14 (undamaged and consolidated assemblies); HI-STORM 100 FSAR Table 6.4.12 and 6.4.13 (bare fuel rod diameter); and HI-STORM 100 FSAR Section 6.4.4.2.3 and Table 6.4.13 (enrichment variations). In all cases the results of those evaluations are equivalent to, or bounded by those for the bare fuel rods arrays. Since the generic approach of modeling damaged fuel and fuel debris is unchanged from the HI-STORM 100, these evaluations are still applicable and need not be re-performed for the HI-STORM FW.

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 undamaged 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 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.

6.4.7 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware as defined in Table 2.1.1 are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies, except for ITTRs, which are placed into the instrument tube.

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 and instrument tubes, i.e., any absorption of the hardware is neglected. Table 6.4.10 shows results for all PWR assembly classes at 5% enrichment with filled and voided guide and instrument tubes in the MPC-37 basket. These results show that for all classes, the condition with filled guide and instrument tubes bound those, or are statistically equivalent to those, with voided guide and instrument tubes. For the higher soluble boron concentration required in the presence of damaged fuel, the same is shown in Table 6.4.5 (two columns on the right). In this case, only the bounding case (Assembly class 15x15F as undamaged fuel) was analyzed.

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.

6.4.8 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM FW 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 k_{eff} 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.

6.4.9 Low Enriched, Channeled BWR fuel

The calculations in this subsection show that low enriched, channeled BWR fuel with indeterminable cladding condition is acceptable for loading in all storage locations of the MPC-89 without placing those fuel assemblies into DFC/DFIs, hence classifying those assemblies as undamaged. The main characteristics that must be assured are:

TABLE 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED EXTERNAL WATER DENSITIES

Water Density		Maximum k_{eff}			
Internal	**External**	**MPC-37 (17x17B, 5.0%)**	**MPC-89 (10x10A, 4.8%)**	**MPC-32ML (16x16D, 5.0%)**	**MPC-44 (14x14B, 5.0%)**
100%	100%	0.9380	0.9435	0.9427	0.9436
100%	70%	0.9377	0.9432	0.9423	0.9433
100%	50%	0.9399	0.9439	0.9429	0.9434
100%	20%	0.9366	0.9428	0.9425	0.9433
100%	10%	0.9374	0.9437	0.9426	0.9433
100%	5%	0.9376	0.9435	0.9426	0.9432
100%	1%	0.9383	0.9435	0.9429	0.9430

TABLE 6.4.2

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

MPC-37 (17x17B, 5.0% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9175	0.8306
50	0.9325	0.9093
75	0.9357	0.9349
100	0.9380	0.9380
MPC-89 (10x10A, 4.8% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9204	0.8345
50	0.9382	0.9128
75	0.9416	0.9392
100	0.9435	0.9435
MPC-32ML (16x16D, 5.0% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9203	0.8213
50	0.9374	0.9175
75	0.9411	0.9399
100	0.9427	0.9427
MPC-44 (14x14B, 5.0% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9195	0.8337
50	0.9373	0.9198
75	0.9420	0.9390
100	0.9436	0.9436

TABLE 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Case	Maximum k_{eff}	
	Dry	**Flooded with unborated water**
MPC-37 (17x17B, 5.0% Enrichment)	0.9335	0.9380
MPC-89 (10x10A, 4.8% Enrichment)	0.9391	0.9435
MPC-32ML (16x16D, 5.0% Enrichment)	0.9361	0.9427
MPC-44 (14x14B, 5.0% Enrichment)	0.9400	0.9436

TABLE 6.4.4

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Maximum k_{eff}	
	Preferential Flooding	**Fully Flooded**
MPC-37 with 12 DFCs (5% Enrichment, Undamaged assembly 15x15F, 20x20 Bare Rod Array)	0.8705	0.9276
MPC-89 with 16 DFCs (4.8 % Enrichment, Undamaged assembly 10x10A, 9x9 Bare Rod Array)	0.8296	0.9464
MPC-32ML with 8 DFCs (5% Enrichment, Undamaged assembly 16x16D, 22x22 Bare Rod Array)	0.8667	0.9380
MPC-44 with 12 DFCs (5% Enrichment, Undamaged assembly 14x14B, 18x18 Bare Rod Array)	0.8460	0.9384

TABLE 6.4.5

MAXIMUM k_{eff} VALUES WITH REDUCED WATER DENSITIES

Internal Water Density† in g/cm³	**Maximum k_{eff}**										
	MPC-89 10x10A, 4.8%	**MPC-37 (1500ppm) 17x17B, 4.0 %**		**MPC-37 (2000ppm) 17x17B, 5.0 %**		**MPC-37† (2300ppm) 15x15F and Damaged Fuel 5.0 %**		**MPC-32ML (2000ppm) 16x16D, 5.0 %**		**MPC-44 (1900ppm) 14x14B, 5.0 %**	
Guide Tubes	N/A	filled	void	filled	void	filled	void	filled	void	filled	void
1.00	0.9435	0.9181	0.9071	0.9380	0.9292	0.9276	0.9265	0.9427	0.9397	0.9436	0.9376
0.99	0.9415	0.9181	0.9059	0.9367	0.9296	0.9271	0.9264	0.9420	0.9389	0.9429	0.9370
0.98	0.9391	0.9162	0.9054	0.9368	0.9279	0.9271	0.9257	0.9423	0.9376	0.9418	0.9359
0.97	0.9370	0.9166	0.9035	0.9364	0.9272	0.9265	0.9242	0.9422	0.9373	0.9417	0.9342
0.96	0.9345	0.9147	0.9005	0.9360	0.9265	0.9265	0.9232	0.9408	0.9366	0.9407	0.9342
0.95	0.9304	0.9148	0.9010	0.9356	0.9243	0.9253	0.9217	0.9414	0.9358	0.9401	0.9327
0.94	0.9280	0.9133	0.8995	0.9335	0.9238	0.9255	0.9225	0.9408	0.9346	0.9388	0.9315
0.93	0.9259	0.9128	0.8986	0.9355	0.9237	0.9263	0.9214	0.9423	0.9376	0.9389	0.9299
0.92	0.9232	0.9120	0.8955	0.9327	0.9203	0.9237	0.9204	0.9399	0.9328	0.9385	0.9287
0.91	0.9183	0.9105	0.8947	0.9335	0.9208	0.9229	0.9194	0.9400	0.9321	0.9367	0.9276
0.90	0.9169	0.9090	0.8934	0.9303	0.9189	0.9226	0.9169	0.9387	0.9307	0.9360	0.9265
0.85	0.9013	0.9042	0.8840	0.9272	0.9109	0.9190	0.9127	0.9352	0.9250	0.9306	0.9181
0.80	0.8850	0.8973	0.8733	0.9222	0.9022	0.9138	0.9040	0.9293	0.9169	0.9237	0.9084
0.70	0.8462	0.8813	0.8477	0.9068	0.8780	0.9000	0.8851	0.9140	0.8981	0.9051	0.8845
0.60	0.7980	0.8565	0.8132	0.8866	0.8478	0.8806	0.8571	0.8902	0.8695	0.8780	0.8532
0.40	0.6762	0.7876	0.7195	0.8244	0.7585	0.8192	0.7735	0.8082	0.7844	0.7899	0.7614
0.20	0.5268	0.6827	0.5806	0.7284	0.6298	0.7237	0.6517	0.6691	0.6510	0.6517	0.6331
0.10	0.4649	0.6206	0.5112	0.6698	0.5639	0.6669	0.5889	0.5855	0.5775	0.5778	0.5702

† External moderator is modeled at 100%.

† With undamaged and damaged fuel. All other cases with undamaged fuel only

TABLE 6.4.11

MAXIMUM k_{eff} VALUES IN THE MPC-32ML WITH UNDAMAGED (16x16D) AND DAMAGED FUEL

Bare Rod Array inside the DFC	Maximum k_{eff}, 4.0 wt%	Maximum k_{eff}, 5.0 wt%
17x17	0.9158	0.9337
19x19	0.9178	0.9367
20x20	0.9182	0.9377
21x21	0.9186	0.9378
22x22	0.9185	0.9380
23x23	0.9181	0.9377
24x24	0.9183	0.9367
26x26	0.9164	0.9356

TABLE 6.4.12

MAXIMUM k_{eff} VALUES IN THE MPC-44 WITH UNDAMAGED (14x14B) AND DAMAGED FUEL

Bare Rod Array inside the DFC	**Maximum k_{eff}, 4.0 wt%**	**Maximum k_{eff}, 5.0 wt%**
14x14	0.9129	0.9340
15x15	0.9145	0.9359
16x16	0.9151	0.9362
17x17	0.9155	0.9382
18x18	0.9159	0.9384
19x19	0.9153	0.9378
20x20	0.9160	0.9381
21x21	0.9135	0.9367
22x22	0.9133	0.9363
23x23	0.9126	0.9351
24x24	0.9128	0.9346
26x26	0.9126	0.9341

APPENDIX 6.B: MISCELLANEOUS INFORMATION

6.B.1 Sample Input File MPC-37

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

6.B.2 Sample Input File MPC-89

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

6.B.3 Analyzed Distributed Enrichment Patterns

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

6.B.4 Assembly Cross Sections

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

6.B.5 Sample Input File MPC-32ML

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

6.B.6 Sample Input File MPC-44

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

CHAPTER 9: OPERATING PROCEDURES

9.0 INTRODUCTION

This chapter contains the operating procedures required for the dry storage of spent nuclear fuel at an on-site HI-STORM FW ISFSI. The decay heat, initial enrichment, burnup and cooling time of the SNF must accord with the restrictions in the Technical Specification. The unloading procedure is also described in this chapter. This sequence of activities is collectively referred to as short-term operations in this safety analysis report (SAR).

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 are met and the requirements of the Certificate of Compliance (CoC) are complied with *literally*. The information provided in this chapter complies with the provisions of NUREG-1536 [9.0.1].

The information presented in this chapter along with the technical basis of the system design described in this SAR will be used to develop detailed operating procedures. Equipment specific operating details such as valve manipulation, canister drying method, special rigging, etc., will be provided to individual users of the system based on the specific ancillary equipment selected and the configuration of the site. In preparing the site-specific procedures, the user must consult the conditions of the CoC, equipment-specific operating instructions, and the plant's working procedures as well as the information in this chapter to ensure that the short-term operations shall be carried out with utmost safety and ALARA.

The following generic criteria shall be used to determine whether the site-specific operating procedures developed pursuant to the guidance in this chapter are acceptable for use:

- All heavy load handling instructions are in keeping with the guidance in industry standards, and Holtec-provided instructions.
- The procedures are in conformance with this FSAR and the COC.
- The operational steps are ALARA.
- The procedures contain provisions for documenting successful execution of all safety significant steps for archival reference.
- Procedures contain provisions for classroom and hands-on training and for a Holtec-approved personnel qualification process to ensure that all operations personnel are adequately trained.
- The procedures are sufficiently detailed and articulated to enable craft labor to execute them in *literal compliance* with their content.

9.1 TECHNICAL AND SAFETY BASIS FOR LOADING AND UNLOADING PROCEDURES

The procedures herein are developed for the loading, storing, and unloading of spent fuel in the HI-STORM FW system. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present physical risk to the operations staff. The design of the HI-STORM FW system, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize potential risks and mitigate consequences of potential events.

The primary objective of the information presented in this chapter is to identify and describe the sequence of significant operations and actions that are important to safety for cask loading, cask handling, storage operations, and cask unloading to adequately protect health and minimize danger to life or property, protect the fuel from significant damage or degradation, and provide for the safe performance of tasks and operations.

The safety evaluation of the various loading configurations and ancillaries is outside the scope of this FSAR because such equipment and analyses must, of necessity, be site specific to accord with the exigencies of the architecture of each plant. However, to ensure consistency, a series of generic reports [9.1.4 thru 9.1.7] that address various loading scenarios have been adopted in Holtec's configuration control system to standardize the analysis methodologies to the extent possible, where none exists in the plant's existing design basis.

In the event of an extreme abnormal condition the appropriate procedural guidance to respond to the situation must be available and ready for implementation. 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.

While still underwater, a thick shielded lid (the MPC lid) is installed. The lift yoke remotely engages to the HI-TRAC VW lift blocks or to the HI-TRAC VW Version P lifting trunnions to lift the HI-TRAC VW and loaded MPC close to the spent fuel pool surface. When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC VW from the spent fuel pool, the cask is removed from the spent fuel pool. The lift yoke and HI-TRAC VW are decontaminated, in accordance with instructions from the site's radiological protection personnel, as they are removed from the spent fuel pool.

HI-TRAC VW is placed in the designated preparation area and the lift yoke is removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC VW are decontaminated. The neutron shield water jacket is filled with water (if drained). The inflatable annulus seal is removed and an annulus shield is installed. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. For HI-TRAC VW Version V2, following decontamination, HI-TRAC VW is loaded into the Neutron Shield Cylinder (NSC) Assembly in the preparation area and attached to the NSC via fasteners/bolts.

Note:
HI-TRAC VW Version V2 contains a NSC Assembly for neutron shielding in lieu of a water jacket. The NSC contains Holtite-A for neutron shielding. Therefore operations steps involving a water jacket are not applicable to the HI-TRAC VW Version V2.

The MPC water level and annulus water level are lowered slightly, the MPC is vented, and the MPC lid is welded on using the automated welding system. Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. A progressive PT examination as described in the Code Alternatives listed in the CoC 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 MPC welds are then pressure tested~~ followed by an additional liquid penetrant examination 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 measurement or determined analytically. The remaining bulk water in the MPC is drained.

Caution:
Inert gas must be used any time the fuel is not covered with water to prevent oxidation of the fuel cladding. The fuel cladding is not to be exposed to air at any time during loading operations.

Depending on the burn-up or decay heat load of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system (VDS) or forced helium dehydration (FHD) system. For MPCs without high burn-up fuel or with high burnup fuel and with sufficiently low decay heat, the vacuum drying system may be connected to the MPC and

used to remove all liquid water from the MPC. For MPCs with high burn-up fuel and higher heat loads, cyclic vacuum drying may be performed in accordance with Chapter 4 of this FSAR and ISG-11 Rev. 3. The annular gap between the MPC and HI-TRAC is filled with water during vacuum drying to promote heat transfer from the MPC and maintain lower fuel cladding temperatures. The internal pressure is reduced and held in accordance with Technical Specifications to ensure that all liquid water is removed.

An FHD system may also be used for high-burn-up fuel at higher decay heat heats as well as for moderate burn-up fuel and HBF at lower heat loads 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 in accordance with Technical Specification requirements to ensure that all liquid water is removed.

Following MPC moisture removal, by VDS or FHD, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, and provides an inert atmosphere for long-term fuel integrity. 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). The cover plate welds are then leak tested.

An option available on all MPCs is the addition of a second cover plate on the drain and vent ports. The outer cover plate is installed in a counterbored recess directly over the inner port cover. The outer port cover is welded with visual and liquid penetrant examinations performed on the root, final, and at least one intermediate weld pass.

The MPC closure ring is then placed on the MPC and 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 MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination. HI-TRAC VW surface dose rates are measured in accordance with the technical specifications. The MPC lift attachments are installed on the MPC lid. The MPC lift attachments are the primary lifting point on the MPC. MPC slings are installed between the MPC lift attachments and the lift yoke.

MPC transfer may be performed inside or outside the fuel building. The empty HI-STORM FW overpack is inspected and positioned with the lid removed. Next, the mating device is positioned on top of the HI-STORM FW and HI-TRAC VW is placed on top of it. The mating device assists in the removal of the HI-TRAC VW bottom lid and helps guide the HI-TRAC VW during its placement on the HI-STORM FW. The MPC slings are attached to the MPC lift attachments. The MPC is transferred using a suitable load handling device.

Note:

If the transfer cask is expected to be operated in an environment below 32 °F, and a minimum heat load requirement was not applied to loading the MPC the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with clean potable or demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Cask weights shall be evaluated to ensure that the equipment load limitations are not violated. (Not applicable for HI-TRAC VW Version V2).

Note (HI-TRAC VW :

HI-TRAC VW Version V2 contains utilizes the NSC Assembly for neutron shielding in lieu of a water jacket. The NSC contains Holtite-A for neutron shielding. Therefore operational~~s~~ steps involving a water jacket are not applicable to the HI-TRAC VW Version V2.

k. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.

l. Disconnect any special rigging from the MPC lid and disengage the lift yoke in accordance with site-approved rigging procedures.

Warning:

MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could be an indication that fuel assemblies not meeting the CoC have been loaded.

m. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.

n. Perform decontamination and a dose rate/contamination survey of HI-TRAC.

o. Prepare the MPC annulus for MPC lid welding by removing the annulus seal and draining the annulus approximately 6 inches.

2. Prepare for MPC lid welding as follows:

a. Clean the vent and drain ports to remove any dirt or standing water. Install the RVOAs to the MPC lid vent and drain ports, leaving caps open.

b. Lower the MPC internal water level in preparation for MPC lid-to-shell welding.

ALARA Note:

The MPC exterior shell survey is performed. 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.

c. Survey the MPC lid top surfaces and the accessible areas (approximately the top three inches) of the MPC external shell. Decontaminate the MPC lid and accessible surfaces of the MPC shell in accordance with LCO 3.2.1.

3. Weld the MPC lid as follows:

   a. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform and to close the gap to the requirements of the licensing drawings.

   b. Install the Automated Welding System (AWS).

Note:
It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.
Caution:
A radiolysis of water may occur in high flux conditions inside the MPC creating combustible gases. 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 purged with inert gas prior to, and during MPC lid welding operations, including welding, grinding, and other hot work, to provide additional assurance that flammable gas concentrations will not develop in this space.

   c. Perform combustible gas monitoring and purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.

Note:
MPC closure welding procedures dictate the performance requirements and acceptance requirements of the weld examinations.

   d. Perform the MPC lid-to-shell weld and NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.

4. Perform MPC lid-to-shell weld pressure testing in accordance with site-approved procedures.

5. ~~Repeat the liquid penetrant examination on the final pass of the MPC lid-to-shell weld~~Examine the MPC for leakage..

   a. In the event of leakage, ~~R~~repair any weld defects in accordance with the applicable code requirements and re-perform the NDE in accordance with approved procedures.

Note:
The demoisturizer module must maintain the temperature of the helium exiting the FHD below the Technical Specification limits continuously from the end of the drying operations until the MPC has been backfilled and isolated. If the temperature of the gas exiting the FHD exceeds the temperature limit, the dryness test must be repeated and the backfill re-performed.

e. Continue operation of the FHD system with the demoisturizer on.

f. 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 LCO 3.1.1.

g. Open the FHD bypass line and Close the vent and drain port RVOAs.

Warning:

A HI-TRAC VW Version V or V2 containing an MPC loaded with spent fuel assemblies shall NOT be left unattended to ensure that blockage of the air flow paths does not occur. The HI-TRAC vents shall be monitored to be free from blockage once every 4 hours.

h. For HI-TRAC VW Versions V and V2, deflate the lower inflatable annulus seal and remove the annulus shield to establish air flow through the annulus.

i. Shutdown the FHD system and disconnect it from the RVOAs.

j. Remove the vent and drain port RVOAs.

9. Weld the vent and drain port cover plates and perform NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable ~~code~~ Code and re-perform the NDE until the weld meets the required acceptance criteria.

If using redundant port cover plates, install the redundant port cover plate, perform the multi-pass welds, and perform NDE on the redundant port cover plates with approved procedures (See 9.1 and Table 2.2.14). Repair any weld defects in accordance with the site's approved Code weld repair procedures.

~~9.~~10. If not using a redundant port cover plate, ~~P~~perform a leakage test of the MPC vent port cover plate and drain port cover plate in accordance with the following and site-approved procedures:

a. If necessary, remove the cover plate set screws or plugs.

b. Flush the cavity with helium to remove the air and immediately install the set screws or plugs recessed below flush of the top of the cover plate.

c. Plug weld the recess above each set screw or plug to complete the penetration closure welding in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable Code and re-perform the NDE until the weld meets the required acceptance criteria.

d. Flush the area around the vent and drain cover plates with compressed air or nitrogen to remove any residual helium gas.

e. Perform a helium leakage rate test of vent and drain cover plate welds in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and leakage test methods and procedures of ANSI N14.5 [9.1.2]. The MPC Helium Leak Rate acceptance criterion is provided in LCO 3.1.1.

~~10.~~11. Weld the MPC closure ring as follows:

a. Install and align the closure ring.

b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.

c. If necessary, remove the AWS.

9.2.5 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 VW are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to SAR Chapter 4.

1. Drain the remaining water from the annulus.

2. Perform the HI-TRAC VW surface dose rate measurements in accordance with the Technical Specifications. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10CFR 72.104(a). Remove any surface contamination from the HI-TRAC surfaces as required by LCO 3.2.1.

Note: HI-STORM FW receipt inspection and preparation may be performed independent of procedural sequence, but prior to transfer of the loaded MPC. See Table 9.2.3 for example of HI-STORM FW Receipt Inspection Checklist.

left in place.

Note:
For HI-TRAC VW Version V2, the HI-TRAC VW is unfastened and removed from the NSC Assembly only after filling of the MPC with water (borated as required).

The top surfaces of the HI-TRAC VW and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC VW lift blocks (for HI-TRAC VW Version P, the lift yoke is engaged to the HI-TRAC VW lifting trunnions). If weight limitations require, the neutron shield jacket is drained of water or the NSC is unbolted from the HI-TRAC. HI-TRAC VW is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are cleared of any assembly debris and crud. HI-TRAC VW and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and overpack are decontaminated.

9.4.2 HI-STORM FW Recovery from Storage

1. Recover the MPC from HI-STORM FW as follows:

    a. Perform a transport route walkdown to ensure that the cask transport conditions are met.

    b. Transfer HI-STORM FW to the fuel building or site designated location for the MPC transfer.

    c. Position HI-STORM FW under the lifting device.

    d. Remove the HI-STORM FW lid.

    e. Install the mating device with bottom lid on top of the HI-STORM FW.

    f. Remove the MPC lift attachment plugs and install the MPC lift rigging to the MPC lid.

2. At the site's discretion, perform a HI-TRAC VW receipt inspection and cleanliness verification in accordance with a site-specific inspection checklist.

Note:
If the HI-TRAC VW is expected to be operated in an environment below 32 ^{o}F, and a minimum heat load requirement was not applied to loading the MPC the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

9.5 REFERENCES

[9.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[9.1.1] U.S. Code of Federal Regulations, Title 10 "Energy", Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,"

[9.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

[9.1.3] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code".

[9.1.4] Holtec Report HI-2135841, "Methodology for Determining Cask-to-ISFSI Pad Dynamic Response with Consideration of SSI and Nonlinear Effects", Latest Revision.

[9.1.5] Holtec Report HI-2135869, "Site-Specific Tornado Missile Analysis for HI-STORM FW System", Latest Revision.

[9.1.6] Holtec Report HI-2146258, "VCT Stability Analysis on Haul Path and ISFSI Pad for Multiple Nuclear Power Plants", Latest Revision.

[9.1.7] Holtec Report HI-2167059, "Analysis Methodology for ANSI N14.6 Special Lifting Devices", Latest Revision.

[9.5.1] U.S. Code of Federal Regulations, Title 10 " Energy", Part 20, "Standards for Protection Against Radiation,"

10.1.2.2 Pressure Testing

10.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

All HI-TRAC transfer cask water jackets shall be hydrostatically tested in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

10.1.2.2.2 MPC Confinement Boundary

Pressure testing (hydrostatic or pneumatic) of the MPC Confinement Boundary shall be performed to verify the lid-to-shell field weld in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 and applicable sub-articles, when field welding of the MPC lid-to-shell weld is completed. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure. The calibrated test pressure gage installed on the MPC Confinement Boundary shall have an upper limit of approximately twice that of the test pressure. Digital type pressure gages may be used without conforming to the upper limit restriction, provided that the combined error due to calibration and readability does not exceed 1% of the test pressure. The MPC vent and drain ports will be used for pressurizing the MPC cavity. Water shall be pumped into the MPC drain port until water only is flowing from the MPC vent port. The MPC vent port is then closed and the pressure is increased to the test pressure. While the MPC is under pressure, the MPC lid-to-shell weld shall be examined for leakage. If any leaks are observed, the pressure shall be released and the weld shall be repaired in accordance with the requirements of ASME Code, Section III, Subsection NB. Following completion of the required hold period at the test pressure, ~~the pressure shall be released and~~ the

surface of the MPC lid-to-shell weld shall be visually re-examined for leakage. ~~by liquid penetrant examination in accordance with ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria.~~ Any evidence of leakage, cracking or deformation shall be cause for rejection, or repair and retesting, as applicable.

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

10.1.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Certain versions of the HI-TRAC include Holtite neutron shielding material.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested are identified in Table 3.1.9 and applicable weld materials. Table 3.1.9 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

For Holtite neutron shielding material, each manufactured lot of material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet the requirements specified in Table 1.2.5. Appendix 1.B of HI-STORM 100 System FSAR [1.1.3] provides the Holtite-A material properties germane to its function as a neutron shield. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration, and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

10.1.4 Leakage Testing

Leakage testing shall be performed in accordance with written and approved procedures and the leakage test methods and procedures of ANSI N14.5 [10.1.5], as follows.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The acceptance criterion is "leaktight" as defined in ANSI N14.5. Shop leakage tests of the base metals and enclosure welds may be performed using automated leak test equipment to minimize the need for operator actions and interpretations. Automated leak test equipment design and computer software programs shall be reviewed and approved by a Level III Leak Test specialist qualified in accordance with ANSI N14.5. Maintenance and calibration of the equipment and testing of the software shall be performed by individuals qualified in accordance with ANSI.N14.5 using written procedures produced under the licensee's quality program. The placement of the MPC components in the test equipment and recording of the test data in the documentation package for the equipment shall be performed by personnel trained and qualified in accordance with the licensee's quality program. The helium leakage test of the vent and drain port cover plate welds shall be performed using a helium mass spectrometer leak detector (MSLD). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

An option available on all MPCs is the addition of a second cover plate on the drain and vent ports. The outer cover plate is installed in a counterbored recess directly over the inner port cover. The outer port cover is welded using a minimum of three weld passes that bridge the weld joint. Visual and liquid penetrant examinations shall be performed on the root, final and at least one intermediate weld pass. Helium leak testing is not required when the redundant port cover design is used

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are not required. Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

Leakage testing of the vent and drain port cover plate welds when required shall be performed after welding of the cover plates and subsequent NDE. For instances where redundant port covers have been installed, leakage testing is not required. The description and procedures for these field leakage tests are provided in Chapter 9 of this FSAR and the acceptance criteria are defined in the Technical Specifications for the HI-STORM FW system.

10.1.5 Component Tests

10.1.5.1 Valves, Pressure Relief Devices, and Fluid Transport Devices

There are no fluid transport devices associated with the HI-STORM FW system. The only valve-like components in the HI-STORM FW system are the specially designed caps installed in the

Table 10.1.1
MPC INSPECTION AND ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of MPC code welds per ASME Code Section III, Subsection NB, as defined on design drawings, per NB-5300, as applicable. b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements. c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements. d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. Acceptance criteria for non-code welds are defined on the drawings. e) Cleanliness of the MPC shall be verified upon completion of fabrication. f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.	a) The MPC shall be visually inspected prior to placement in service at the licensee's facility. b) MPC protection at the licensee's facility shall be verified. c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.	a) None.

Table 10.1.1 (continued) MPC INSPECTION AND ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Structural	a) Assembly and welding of MPC components is performed per ASME Code Section IX and III, Subsection NB, as applicable. b) Materials analysis (steel, neutron absorber, etc.), is performed and records are kept in a manner commensurate with "important to safety" classifications.	a) None.	a) A multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld is performed per ASME Section V, Article 2. Acceptance criteria for the examination are defined in Subsection 10.1.1, and in the Licensing Drawings. b) ASME Code NB-6000 pressure test is performed after MPC closure welding. Acceptance criteria are defined in the Code.
Leak Tests	a) Helium leakage testing of the MPC base metal (shell, baseplate and MPC lid),MPC shell to baseplate welds and MPC shell to shell welds is performed on the unloaded MPC. Acceptance criterion is in accordance with "leaktight" definition in ANSI N14.5.	a) None.	a) Helium leakage testing is performed on the vent and drain port cover plates to MPC lid field welds and the cover plate base metals.~~,~~ If the redundant port cover design is used on the vent and drain ports, helium leak testing is not required. See Technical Specification for guidance on acceptance criteria.
Criticality Safety	a) The boron content is verified at the time of neutron absorber material manufacture. b) The installation of MPC cell panels is verified.	None.	None.
Shielding Integrity	a) Material compliance is verified through CMTRs. b) Dimensional inspection of MPC lid thickness is performed.	None.	None.

Table 10.1.1 (continued)
MPC INSPECTION AND ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a)None.	a) None.	a) None.
Fit-Up Verification	a) Fit-up of the following components is verified during fabrication. - MPC lid - vent/drain port cover plates - MPC closure ring b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components is verified during pre-operation. -MPC lid -MPC closure ring -vent/drain cover plates	a) None.
Canister Identification Verification	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 10.1.4
HI-STORM FW ~~MPC~~ NDE REQUIREMENTS

Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
Shell longitudinal seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Lid-to-shell	PT (root and final pass) and multi-layer PT. PT (surface following pressure test)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid (Single port cover plate option) ~~Port cover plates-to-lid (when in conjunction with redundant port cover plate)~~	PT (root and final pass) ~~PT (final pass)~~	ASME Section V, Article 6 (PT) ~~ASME Section V, Article 6 (PT)~~	PT: ASME Section III, Subsection NB, Article NB-5350 ~~PT : Clean White~~
Port cover plates-to-lid (Redundant Port ~~C~~cover ~~p~~Plates ~~to lid~~ Option)	Inner Plate: PT (root and final pass) Outer Plate: PT (root, final and at least one intermediate pass)	ASME Section V, Article 6 (PT	PT: ASME Section III, Subsection NB, Article NB-5350. In addition, the PT of the Inner Plate shall produce a clean, "White" result to indicate a lack of porosity
Lift lug and lift lug baseplate	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350

CHAPTER 12†: ACCIDENT ANALYSIS

12.0 INTRODUCTION

This chapter presents the evaluation of the HI-STORM FW System for the effects of off-normal and postulated accident conditions; and other scenarios that warrant safety analysis (such as MPC reflood during fuel unloading operations), pursuant to the guidelines in NUREG-1536. The design basis off-normal and postulated accident events, including those based on non-mechanistic postulation as well as those caused by natural phenomena, are identified. For each postulated event, the event cause, means of detection, consequences, and corrective actions are discussed and evaluated. For other miscellaneous events (i.e., those not categorized as either design basis off-normal or accident condition events), a similar outline for safety analysis is followed. As applicable, the evaluation of consequences includes the impact on the structural, thermal, shielding, criticality, confinement, and radiation protection performance of the HI-STORM FW System due to each postulated event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM FW System under the short-term operations and various conditions of storage are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and analyses reported therein.

It should be noted that HI-TRAC VW the evaluations provided in this chapter are for HI-TRAC VW with a water jacket. These evaluations are also valid for HI-TRAC VW Version V2 which uses a neutron shield cylinder (instead of water jacket), which employs Holtite-A for neutron shielding-, except the water jacket and water in water jacket are replaced by neutron shield cylinder and Holtite in lieu of water jacket and water. Because of this similarity and for clarity purposes, HI-TRAC VW Version V2 is not explicitly discussed further in this chapter.

Chapter 12 is in full compliance with NUREG-1536; no exceptions are taken.

† 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.

System malfunction event is evaluated assuming the following bounding conditions:

a. Steady state maximum temperatures have been reached
b. Design maximum heat load in the limiting MPC~~-37~~
c. Air in the HI-TRAC VW annulus
d. The helium pressure in the MPC is at the minimum possible value from the technical specification.

The results of a steady state analysis (which implies an extended period of FHD unavailability) are provided in Section 4.6. The results provide the assurance that the peak fuel cladding temperature in the MPC will remain below the ISG-11 limit (see Table 2.2.3) in the event of a prolonged unavailability of the FHD system under the most thermally adverse conditions (highest possible heat load absence of any forced heat removal measures and minimum system helium pressure).

iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

iv. Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

v. 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 MPC structural boundary internal pressures cannot exceed the normal condition pressure limits, assuring Confinement Boundary integrity.

vi. 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 FW System.

12.1.5.4 Corrective Action for FHD Malfunction

The HI-STORM FW System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently, no corrective action is required.

CHAPTER 13†: OPERATING CONTROLS AND LIMITS

13.0 INTRODUCTION

The HI-STORM FW system provides passive dry storage of spent fuel assemblies in interchangeable MPCs with redundant multi-pass welded closure. The loaded MPC is enclosed in a single-purpose ventilated or unventilated metal-concrete overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STORM FW system at an ISFSI. The information provided in this chapter is in full compliance with NUREG-1536 [13.1.1].

13.1 PROPOSED OPERATING CONTROLS AND LIMITS

13.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

This portion of the FSAR establishes the commitments regarding the HI-STORM FW system and its use. Other 10CFR72 [13.1.2] and 10CFR20 [13.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [13.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM FW system. The general license conditions governed by 10CFR72 [13.1.2] are not repeated within these Technical Specifications. Licensees are required to comply with all commitments and requirements.

The Technical Specifications provided in Appendix A to the CoC and the authorized contents and design features provided in Appendix B to the CoC are primarily established to maintain subcriticality, the confinement boundary, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 13.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 13.1.2 provides the list of Technical Specifications for the HI-STORM FW system.

† 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.

13.2.3 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 FW 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 (not applicable for the unventilated Overpack).

13.2.4 Limiting Conditions for Operation (LCO)

Limiting Conditions for Operation (LCO) specify the minimum capability or level of performance that is required to assure that the HI-STORM FW system can fulfill its safety functions.

13.2.5 Equipment

The HI-STORM FW 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 FW 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.

13.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM FW system fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Subsection 2.1.8 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

13.2.7 Design Features

This subsection describes HI-STORM FW 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 the CoC, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM FW system is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 10.

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SURVEILLANCE REQUIREMENTS (continued)	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. The conditions and requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high), heat load, and the applicable short-term temperature limit are given in the CoC/TS Appendix A, Table 3-1. 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. Meeting the helium leak rate limit prior to TRANSPORT OPERATIONS ensures there is adequate helium in the MPC for long term storage and that there is no credible effluent dose from the MPC. MPCs that utilize the redundant port cover design exhibit increased confinement boundary reliability. Each port cover plate is subjected to NDE to ensure absence of porosity in the material and is welded to the MPC lid in the same manner as in the non-redundant design. Each cover plate weld is subjected to similar NDE acceptance criteria, where successful NDE will verify the associated weld's integrity to maintain the MPC confinement boundary. As such, this surveillance does not need to be performed for MPCs that utilize the redundant port cover design. All 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 MPC design.
REFERENCES	1. FSAR Chapters 1, 4, 7 and 9 2. Interim Staff Guidance Document 11, Rev. 3 3. Interim Staff Guidance Document 18, Rev. 1

BASES	
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). This LCO is not intended to address low frequency, unexpected Design Event III and IV class events (ANSI/ANS-57.9) 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 3.4.10 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. The LCO is not applicable for the unventilated OVERPACK.
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.

(continued)

B 3.3 SFSC Criticality Control

B 3.3.1 Boron Concentration

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 Certificate of Compliance. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the 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 cover plates and MPC closure ring are installed and welded. Inspections are performed on the welds. For those MPCs containing PWR fuel assemblies credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.
APPLICABLE SAFETY ANALYSIS	The spent nuclear fuel stored in the SFSC is required to remain subcritical ($k_{eff} \leq 0.95$) under all conditions of storage. The HI-STORM FW SFSC is analyzed to store a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-37, ~~or~~ MPC-32ML, MPC-37P, or MPC-44, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

(continued)

BASES

LCO

Compliance with this LCO ensures that the stored fuel will remain subcritical with a $k_{eff} \leq 0.95$ while water is in the MPC. LCOs 3.3.1 provides the minimum concentration of soluble boron required in the MPC water for the MPC-37, ~~and~~ MPC-32ML, MPC-37P, and MPC-44. The amount of soluble boron is dependent on the initial enrichment of the fuel assemblies to be loaded in the MPC. Fuel assemblies with an initial enrichment less than or equal to 4.0 wt. % U-235 require less soluble boron than those with initial enrichments greater than 4.0 wt. % U-235. For initial enrichments greater than 4.0 wt. % U-235 and up to 5.0 wt. % U-225, interpolation is permitted to determine the required minimum amount of soluble boron.

All fuel assemblies loaded into the MPC-37, ~~or~~ MPC-32ML, MPC-37P, or MPC-44 are limited by analysis to maximum enrichments of 5.0 wt. % U-235.

The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes are mixed in the same MPC. This ensures the assumptions pertaining to soluble boron used in the criticality analyses are preserved.

APPLICABILITY

The boron concentration LCO is applicable whenever an MPC-37, ~~or~~ MPC-32ML, MPC-37P, or MPC-44 has at least one PWR fuel assembly in a storage location and water in the MPC.

During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water.

During UNLOADING OPERATIONS, the LCO is applicable when the MPC is reflooded with water. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is satisfied upon entering the Applicability.

(continued)

B 5.0 Administrative Controls and Programs (LCO) APPLICABILITY
B 5.3 Radiation Protection Program

BASES	
B 5.3.1	5.3.1 requires that the licensee appropriately includes provisions in their radiation protection program to account for dry storage of the system from loading through unloading. These provisions should also include the requirements included in Section 5.3 of the CoC.
B 5.3.2	5.3.2 includes the requirements of 10CFR72.212(b)(2)(i)(c) for a documented evaluation that the dose limits of 10CFR72.104(a) are met. This evaluation should utilize the site-specific ISFSI layout, the planned number of casks, and the cask contents to demonstrate compliance with 10CFR72.104
B 5.3.3	In accordance with 5.3.3, licensees should use the analysis performed in 5.3.2 to establish individual cask surface dose rate limits for the TRANSFER CASK and the OVERPACK, in accordance with the measurement locations specified in 5.3.8. At the top of the OVERPACK, the side of the OVERPACK, the side of the TRANSFER CASK, and the inlet and outlet ducts on the OVERPACK (if applicable) or on the side of the lid for the unventilated OVERPACK. If measured dose rates exceed these limits, it could be an indication of a loading error that may require corrective actions These calculated limits are used in comparison with the measured values in 5.3.8.
B 5.3.4	5.3.4 establishes maximum dose rates on a loaded OVERPACK or TRANSFER CASK. 5.3.8 establishes locations for surface dose rate measurements. Compliance with 10CFR72.104 dose limits are confirmed with a comparison between these measured dose rates and the dose rate limits of the system set by calculation and maximum limits in 5.3.3 and 5.3.4 as described in 5.3.5. The measurement locations specified in 5.3.8 ensure the measured dose rates are compared with the analysis described in 5.3.3 at the same geometric location. Showing that the calculated dose rates are greater than the measured dose rates at the same location provides assurance that the calculated dose (from 5.3.2) bound the actual doses at the site boundary, and therefore assures compliance with 10CFR72.104(a). The dose rate limits defined in 5.3.4 are lower than the calculated bounding dose rates in FSAR Chapter 5. The bounding dose rates in Chapter 5 are based on allowable contents; however, a loading pattern which would result in such

BASES	
	dose rates would not be in accordance with ALARA practices and would not be used. The dose rate limits set forth in 5.3.4 provide the actual bounding limits used for the system.
B 5.3.5	5.3.5 provides the requirement that the licensee measure dose rates at the locations outlined in 5.3.8 and compare them to the lower of the two limits established in Section 5.3.3 or 5.3.4. This ensures that the most conservative limit is used.
B.5.3.6	5.3.6 establishes corrective actions that shall be taken in the event of measured dose rates that exceed the lower of the two limits in Section 5.3.3 or 5.3.4. These corrective actions include verifying that contents were loaded correctly, performing analyses to ensure 10CFR72.104 dose limits are met, and determining the cause of the higher dose rate.
B 5.3.7	5.3.7 states that any evaluation under 5.3.6 that shows that 10CFR72.104 dose rate limits will not be met will prevent the MPC from being installed in the OVERPACK or it will be removed from the OVERPACK. This control ensures that the site continues to meet all regulatory requirements.
B 5.3.8	5.3.8 establishes locations for surface dose rate measurements. Compliance with 10CFR72.104 dose limits are confirmed with a comparison between these measured dose rates and the dose limits of the system set by calculation and maximum limits in 5.3.3 and 5.3.4 as described in 5.3.5. The measurement locations specified in 5.3.8 ensure the measured dose rates are compared with the analysis described in 5.3.2 at the same geometric location. Comparing the calculated dose rates at the same location as the measured dose rates provides assurance that the calculated dose (from 5.3.2) bound the actual doses at the site boundary, and therefore assures compliance with 10CFR72.104(a). Even though comparison of dose rates can occur across any location, the locations chosen in 5.3.8 were based on positions where higher dose rates are expected. Higher dose rates provide better measurements to protect against measurement inaccuracy and the additional actions of 5.3.6 and 5.3.7 for compliance to 10CFR72.104.

CHAPTER 1.I; GENERAL DESCRIPTION: HI-STORM FW SYSTEM WITH UNVENTILATED OVERPACK

1.I.0 General Information:

Supplement I to the HI-STORM FSAR is a series of supplements labelled Chapter n.I to the existing chapter n in the HI-STORM FW FSAR (n can be from 1 to 14). Supplement I is limited to adding the Unventilated overpack labelled "Version UVH", to the HI-STORM FW Canister storage system. Each individual chapter that requires additional safety analysis to qualify the addition of a new overpack model has its own supplement. Specifically, this supplement to the HI-STORM FW FSAR is limited to the safety analysis of a simplified version of the HI-STORM FW system (hereafter abbreviated as "the *Storage System"*) wherein the overpack's inlet and outlet air passages have been removed resulting in a complete cessation of ventilation in the space between the cask cavity and the stored multi-purpose canister (MPC) during the system's operation. The overpack model is referred to as HI-STORM FW Version UVH (the annex UVH is an abbreviation of **u**n-**v**entilated and H stands for **h**igh density concrete) or simply as "Version UVH." Figure 1.I.0.1 shows a cut-away view of the Storage System.

This supplement contains the necessary information and analyses to support the amendment to the certificate-of-compliance issued to the HI-STORM FW Canister Storage system in docket # 72-1032 to serve as a spent nuclear fuel (SNF) dry storage cask under the provisions of 10 CFR 72 [1.0.1][1]. This supplement, like others, in some cases introduces improved safety analysis methods and better articulated acceptance criteria that provide a more robust bases or comprehensive treatment of the physical problem. Upon regulatory acceptance, such updated analysis techniques, where applicable, should be treated as preferred methods for all editions of this FSAR and the corresponding CoCs.

This supplement, prepared pursuant to 10 CFR 72.230, describes the basis for NRC approval and CoC amendment on the HI-STORM FW System under 10 CFR 72, Subpart L to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI) under the general license authorized under 10 CFR 72, Subpart K.

The purpose of this supplement is to provide a general description of the design features and storage capabilities of the storage system consisting of the Version UVH overpack and the MPCs certified for storage therein. This Supplement introduces no new MPC or transfer cask; the only new equipment introduced is the unvented overpack which is illustrated in the Licensing drawing package in Section 1.I.5. Ancillary equipment in the main report are also applicable to this supplement. This supplement is also suitable for incorporation into a site-specific Safety Analysis Report, which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility that is similar in objective and scope.

An MPC (containing either PWR or BWR fuel) is placed inside the HI-STORM FW Version UVH overpack for extended storage. The overpack provides shielding and environmental protection to the MPC. The HI-TRAC VW transfer cask, used in every HI-STORM FW model, is used for MPC transfer and also provides shielding and protection while the MPC is being prepared for storage.

Supplement I is comprised of a number of chapters where safety-relevant information on the HI-STORM FW system containing the Version UVH overpack is needed. There are, however, several chapters that are

[1] Reference to a section, table, Figure or Reference without a Roman Numeral in it means it is in the main report.

not affected by Version UVH and are therefore omitted. The unaffected chapters are listed in Table 1.I.0.1 along with the rationale for their omission.

Because of the extensive nexus between the SSCs introduced in this Supplement and those previously documented in this FSAR, even the chapters that require fresh safety evaluation material have sections within them that do not. Those sections are identified at the beginning of each chapter and the rationale for their omission is given. For this chapter, Table 1.I.0.2 lists the sections that do not require any change and are therefore not repeated.

All chapters in this supplement are identified as n.I, where n is the chapter number in the main report which it supplements. Likewise, sections within a chapter are denoted by n.I.m, where m is the section number in the main chapter to which it applies. Thus, n.I.m.p represents the sequential sub-section p within section n.I.m.

All tables and figures within each chapter are numbered sequentially. Thus, Table n.I.1.1 represents the first table in Section n.I.1

Thus, the presence of I in the reference to a section, table, reference or figure clearly identifies it to belong to Supplement I.

All tables and figures called out in a Section can be found at the end of that Section.

Table 1.I.0.1; HI-STORM FW FSAR Chapters Unaffected by the Inclusion of Version UVH		
Chapter number	**Title**	**Reason for omission**
6	Criticality Evaluation	No new MPC is introduced in this supplement; therefore, there is no change in the criticality safety of the storage system
7	Confinement	There is no change in the MPC confinement system. Therefore, the assertion made in Chapter 7 with regard to the leak tightness of the Confinement system apply.
8	Materials evaluation	No new material is introduced in this Supplement I.
11	Radiation Protection	The radiation protection attributes of the Storage system are improved in the unvented Storage system because the elimination of the inlet and outlet air passages eliminates associated streaming of radiation during both the MPC loading operations and on-the-pad storage. Therefore, the safety conclusions reached in Chapter 11 are applicable in an even greater measure.
14	Quality Assurance Program	The quality assurance program remains unchanged.

Table 1.I.0.2; Sections in Chapter 1 of the FSAR that Remain Applicable to the Safety Evaluation in Supplement I		
Section number	**Title**	**Reason for omission**
1.I.3	Agents and subcontractors	The information in Section 1.3 does not require any correction

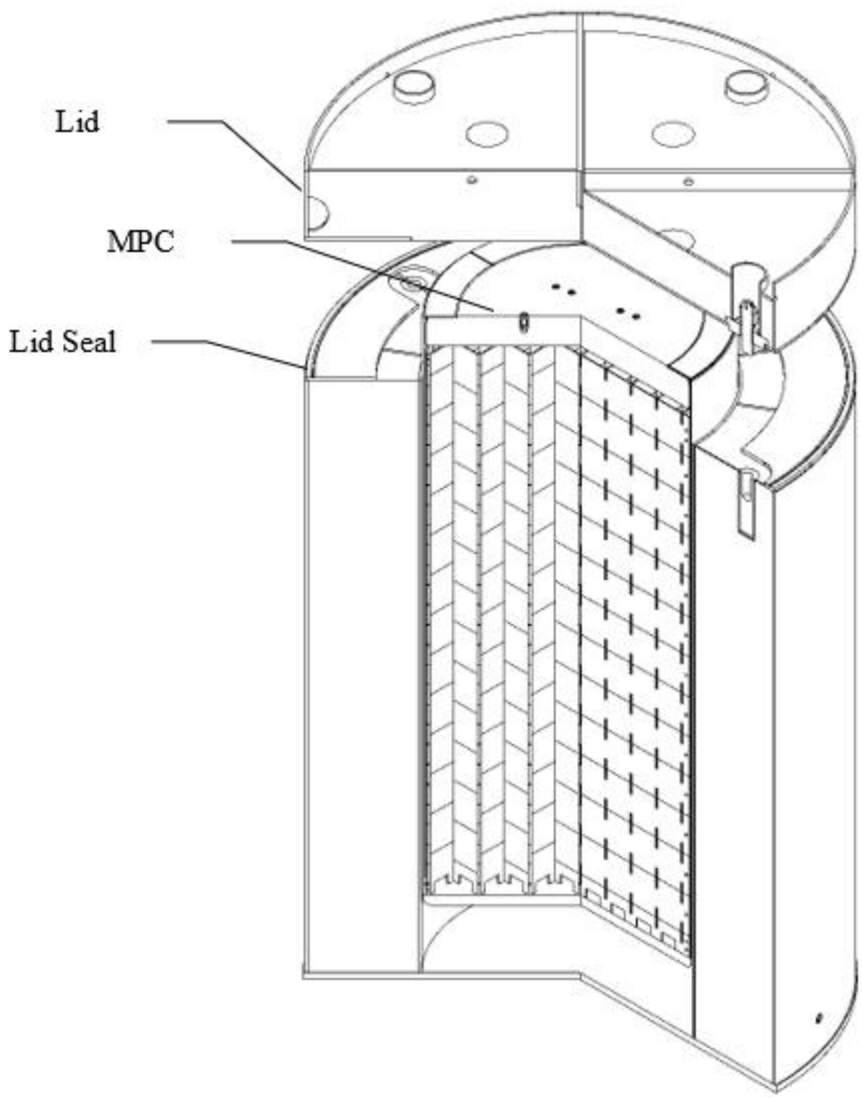

Figure 1.I.0.1 HI-STORM FW Storage System with Version UVH Overpack in cut-away View

1.I.1 Introduction to the Storage System

The HI-STORM FW storage system considered in this supplement is differentiated by the unventilated overpack called Version UVH; all other components in Table 1.I.1.2, namely the MPCs and the HI-TRAC VW transfer cask remain unchanged. To fix ideas, wherever the ventilated HI-STORM FW system is referenced in this Supplement I, it refers to the latest version, namely Version E analyzed and qualified in the main report, without vents and MPC guides.

Because the Storage system does not rely on ventilation action, its heat rejection capacity is rather modest, governed by the natural convection and radiation from the external surfaces of the overpack. To quantify the heat removal rate, a quiescent condition (no wind) is assumed in the thermal analysis summarized in Chapter 4.I.

Version UVH is expected to serve as a low-dose MPC storage system wherein the external environment around the Canister is sought to be controlled, such as to protect from stress corrosion cracking.

In all physical respects, the Storage system is essentially identical to its ventilated counterpart. Thus, like the ventilated HI-STORM FW overpack models, the Version UVH overpack is staged as a free standing configuration on a sheltered or unsheltered pad. Other key characteristics of the Storage system that it shares with other HI-STORM FW systems are:

- Because the cask is not used to load fuel in the pool, the storage system does not run the risk of being infected with the pool's contamination.

- The Canister, designed and qualified to be *leak-tight*, is a compact "*waste package*" which can be readily retrieved and transported off-site in a suitably certified transport cask.

- The MPC confinement boundary, deemed to be leak-tight pursuant to ISG-18, provides an incomparably greater protection against leakage than a gasketed metal cask with a bare basket.

- All SSCs of the UVH listed in Table 1.I.1.1 are designated Important-to-safety (ITS). The ISFSI pad is NITS.

Table 1.I.1.1: Principle System Components QA Designation	
Principle System Components	**QA Designation**
HI-STORM FW UVH Overpack	ITS
HI-TRAC VW Transfer Casks (Table 1.I.1.2)	ITS
MPCs (Table 1.I.1.2)	ITS

Table 1.I.1.2 Principal Components Subject to Certification Associated with the Version UVH in the HI-STORM FW System			
Component I.D.	**Characteristic**	**Function**	**Comment**
MPC-37(Certified in Rev 0 of the CoC)	Storage for 37 PWR fuel assemblies	Provide confinement to its contents under normal, off-normal and accident conditions and during Part 72 Short Term operations	All MPC Fuel Baskets are made of Metamic-HT.
MPC-89 (certified in Rev 0 of the CoC)	Storage for 89 BWR fuel assemblies		
MPC-44 (Certification sought in Rev 7 of the CoC)	Storage for 44 PWR fuel assemblies		
HI-TRAC VW(certified in Rev 0 of the CoC), HI-TRAC VW Version V (certified in Rev 5 of the CoC), HI-TRAC VW Version V2 (certified in Rev 5 of the CoC)	Variable weight transfer cask available in unventilated and ventilated versions	The transfer cask is indispensable to execute Short Term operations.	Version UVH is configured to utilize the same HI-TRAC models as other “FW” overpack models.

1.I.2 General Description

1.I.2.1 System Characteristics

The components of the Storage system are listed in Table 1.I.1.1. The description presented in Section 1.2 of the main FSAR remains applicable for the components listed in Table 1.I.1.1, with only the storage overpack being different. The overpack, illustrated in the licensing drawing in Section 1.I.5, is sized to store the designated reference MPCs listed in Table 1.I.1.2.

1.I.2.1.1 MPCs:

There is no change in the design of MPCs described in Subsection 1.2.1.1 of this FSAR, for applicability to this supplement. This supplement introduces no new MPCs or amends any MPC design.

1.I.2.1.2 Version UVH Overpack:

The HI-STORM FW Version UVH overpack is made from a dual shell steel weldment filled with shielding concrete. Structurally, it emulates a classic metal cask wherein all inlet and outlet vents have been eliminated and the Closure Lid is installed on the cask body, with a weather-resistant metallic gasket providing an enclosed environment in the overpack. Absence of the inlet and outlet air passages and the all-welded steel internal boundary of the overpack enclosed by a steel buttressed and gasketed Closure lid renders the cask's internal space into an environmentally sequestered enclosure. This sealed annular space is envisaged to hold the loaded *multi-purpose canister* in an upright orientation.

As its design configuration would suggest, "Version UVH" (UV is an abbreviation of **unv**entilated, H stands for **h**igh density concrete) has a reduced heat load capacity compared to its ventilated counterpart. Because the only heat rejection pathway available to the Version UVH storage overpack is through natural convection and radiation to the ambient and a limited amount of conduction to the ISFSI pad, the annulus gas inside the overpack will be at an elevated temperature. Because heating of gas reduces its relative humidity and a high humidity content is necessary (but not sufficient) to induce stress corrosion cracking (SCC) in the stainless steel confinement boundary of the MPC, increasing the temperature of the gas surrounding the Canister serves to minimize the incidence of SCC under extended storage conditions. Preventing SCC is a principal objective of Version UVH.

The key distinguishing feature of Version UVH is that it has no inlet or outlet vents. Thus, there is no meaningful ventilation flow of gas around the MPC. Rather the cask is designed to reject the fuel's decay heat from the external surface of the Canister without the benefit of ventilation flow. Rejection of heat from the external surface of the Canister to the external surface of the overpack is facilitated by a combination of conduction and radiation modes of heat transmission. The inside diameter of the overpack has a tight clearance with the OD of the Canister which, under the design basis heat load, computes to an essentially vanishing value giving conduction a bigger role in heat dissipation. Radiation from the hot MPC surfaces to the cask's inner surfaces also plays an active heat dissipation role. Finally, the shielding concrete used in Version UVH is of high density rich in hematite class of aggregate which ensures a high thermal conductance across its mass. Heat rejection from the overpack to the ambient environment like all other HI-STORM overpack models, occurs through natural convection and radiation from the cask's exposed surfaces.

The Closure Lid for Version UVH is also a steel structural weldment with high density, high conductivity concrete installed inside its body to provide protection against sky shine. The Closure Lid is installed on

the cask body by a set of equally spaced anchor bolts with a small clearance and an interposed gasket providing a barrier against intrusion of air in the overpack's annulus space and thus protecting the MPC from the deleterious effect of airborne species that induce stress corrosion cracking (SCC) in stainless steel. Precluding the incidence of SCC in the MPC shell during extended period of storage by creating a still air environment around it is a principal benefit of Version UVH. The weight of the Closure Lid helps the sealing action of the gasket. In the event the air in the overpack annulus were to pressurize, the weight of the lid is counteracted allowing the air to escape. Thus, the overpack has a built-in protection against overpressure.

In addition to providing a barrier against ingress of aggressive species in the space around the MPC, Version UVH also accrues several salutary benefits, such as:

- Absence of vent openings eliminates a source of radiation to the environment emitted from the Canister.

- The overpack is rendered much more rugged against mechanical projectiles in absence of vent openings. The intermediate and penetrant Design Basis Missiles (see Table 2.I.2.1) cease to be a safety concern.

- The Version UVH overpack, made of steel and devoid of any vents, emulates a metal cask in respect of critical functions under accident conditions such as the Design Basis Fire. However, thanks to its larger footprint and greater mass, it is a far superior in respect of shielding capacity and seismic stability in comparison to any peer metal cask.

- The aging related deterioration of the paint on the cask's internal surface is substantially retarded because of the hot and dry environment in contact with it.

- The need for periodic inspection of the vent openings and associated LCOs in the CoC becomes inoperative eliminating this source of radiation dose to the site staff.

Because of the main heat rejection path in Version UVH is conduction through the cask body, a large number of ribs are used to join the inner and outer shells. Likewise, the Closure lid features extensive physical connectivity between its bottom and top surfaces.

In summary, Version UVH overpack emulates a conventional metal cask but provides significantly improved radiation shielding because of its thick and high density concrete filled steel weldment construction. Its other notable characteristics are:

- There is considerable flexibility relative to the height of the cask's internal cavity as well. The cavity should be tall enough to accommodate the tallest MPC that will be stored at the site.

- The density of the shielding concrete can be set at the value needed (between 200 and 250 pcf) to realize the level of dose reduction required.

1.I.2.1.3 Transfer Cask

No new transfer cask design is proposed in this supplement and existing design described in Subsection 1.2.1.3 is not modified.

1.I.2.1.4 Shielding and Neutron Absorber Materials:

There is no change in the materials employed in the HI-STORM FW system with Version UVH overpack.

1.I.2.1.5 Lifting Devices:

There is no change in the specification for the Lifting Devices described in the HI-STORM FW system as described in subsection 1.2.1.5 in the main report.

1.I.2.1.7 Design Life:

There is no change in the design life of the HI-STORM FW system as described in the main body of this FSAR.

1.I.2.2 Operational Features:

The operational features remain fully applicable except that, as stated in Chapter 9.I, before installing the Closure Lid on the Storage overpack, a gasket to inhibit exchange of the gas inside and outside of the cask is placed on the interface between the cask body and the Closure Lid.

1.I.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.I.2.2.3.1 Criticality Prevention

There is no change in the MPCs, and their Fuel Baskets proposed in this Supplement. Therefore, there is no change in the criticality safety characteristics of the Storage system.

1.I.2.2.3.2 Chemical Safety

As stated in 1.2.2.3.2, there are no chemical safety hazards associated with operations of the Storage system. A detailed evaluation is provided in Section 3.4.

1.I.2.2.3.3 Operation Shutdown Modes

The Storage system is totally passive and consequently, operation shutdown modes are unnecessary.

1.I. 2.2.3.4 Instrumentation

As stated in 1.2.2.3.4, the HI-STORM FW MPC, which is seal welded, non-destructively examined, and pressure tested, confines the radioactive contents. The Storage system is completely passive with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, temperature elements may be utilized to monitor the surface temperature of the cask and the temperature of the annulus air.

1.I.2.2.3.5 Maintenance Technique

As stated in 1.2.2.3.5, because of its passive nature, the Storage system requires minimal maintenance over its lifetime. No special maintenance program is required. Supplement 10.I describes the maintenance program specific to the HI-STORM Version UVH.

1.I.2.3 Cask Contents:

The same fuel types allowed in the main chapter, subsection 1.2.3 are allowed for storage in the HI-STORM UVH overpack. However, additional restrictions on heat load apply and are described in Section 2.I.1.1.

1.I.4 Generic Cask Arrays

The discussion in Section 1.4 remains applicable to the Version UVH system with the exception of the allowable pitch between any two adjacent casks. See Table 1.I.4.1 for additional details.

Table 1.I.4.1
Cask Layout Pitch Data

Cask Array	Minimum Allowable Pitch between Adjacent Casks (ft)
2 x N	16
Square Array	

1.I.5 Licensing Drawings:

The licensing drawing package for "Version UVH" is provided in this section.

Drawing Number	Title	Revision
11897	HI-STORM FW Version UVH Licensing Drawing	1

[PROPRIETARY DRAWINGS WITHHELD PER 10CFR2.390]

CHAPTER 2.I: PRINCIPAL DESIGN CRITERIA

2.I.0 Introduction:

The principal design criteria for the Version UVH equipped HI-STORM FW Canister storage system is unchanged in all respects except for those relating to environment control and annulus overpack pressure.

The Version UVH overpack does not have any open penetrations such as air vents in the classical design to permit ventilation of the ambient air. All open vents are eliminated, and the Closure Lid is installed with a concentric gasket which inhibits the exchange of the gas inside the cask with the ambient air. The air in the cask cavity space is filled to a sub-atmospheric pressure to ensure that the internal pressure during operating conditions will always remain below the external ambient pressure, precluding any release of the cavity gas into the ambient. The Closure Lid is emplaced on the cask body with body bolts which are installed with a small axial clearance to allow any significant increase in internal gas pressure above the ambient, under hypothetical scenarios, to be relieved once it overcomes the counteracting lid's weight. A simple force equilibrium shows that a pressure rise of 5 psi in the cask cavity is not possible to sustain even under the scenario of maximum density concrete installed in the cask's lid. However, the structural evaluations are performed by conservatively assuming that the internal pressure is not relieved under hypothetical accident conditions.

The loadings associated with Version UVH must include internal pressure and external pressure which are not present in the ventilated cask. For all other Design Basis Loadings, Version UVH cask body is the same as the standard FW or Version E cask body. In this chapter, the Design pressures appropriate to Version UVH are defined and the overpack loadings are re-visited to ensure that the safety analyses presented in other chapters are comprehensive.

2.I.0.1 Principal Design Criteria for the ISFSI Pad

The principal design criteria for the ISFSI pad applicable for the Version UVH cask remains unchanged from the main body of the FSAR (Table 2.2.9) with the exception of the requirements identified in Table 2.I.0.1.

Table 2.I.0.1

ISFSI Pad Requirements for Version UVH System

Item	Maximum Allowable Value
Concrete Pad Compressive Strength[1] (psi)	6,000

Notes:
[1] Compressive strength of concrete shall be determined based on 28-day break results, consistent with the guidance in NUREG-2215.

2.I.1 Spent Fuel to be Stored

The Version UVH overpack is compatible with select MPC models. All fuel assembly array/classes and non-fuel hardware which are authorized for storage in these MPCs are authorized for storage in Version UVH. All fuel storage characteristics applicable to these MPCs remain unchanged for the Version UVH. See Table 2.I.1.1 for additional information.

However, the permissible heat load is reduced to accord with the diminished heat rejection capacity of Version UVH. This is considered in Chapter 4.I and described in Subsection 2.I.1.1.

2.I.1.1 Design Heat Load

The permissible heat load is also reduced due to the diminished heat rejection capacity of the Version UVH overpack. Permissible heat loads based on MPC type are listed in Tables 2.I.1.2 through 2.IV.1.6.

For PWR fuel with a longer active fuel length than the reference fuel, the maximum total heat load, maximum section heat load limits, and specific heat load limits in each cell, may be increased by the ratio SQRT(L/144), where L is the active length of the fuel in inches.

For PWR fuel with a shorter active fuel length than the reference fuel, the maximum total heat load, maximum section heat load limits, and specific heat load limits in each cell, shall be reduced linearly by the ratio L/144, where L is the active fuel length of the fuel in inches.

For BWR fuel with a longer active fuel length than the reference fuel, the maximum total heat load, maximum section heat load limits, and specific heat load limits in each cell, may be increased by the ratio SQRT(L/150), where L is the active length of the fuel in inches.

For BWR fuel with a shorter active fuel length than the reference fuel, the maximum total heat load, maximum section heat load limits, and specific heat load limits in each cell, shall be reduced linearly by the ration L/150, where L is the active fuel length of the fuel in inches.

2.I.1.2 Radiological Parameters for Spent Fuel and Non-fuel Hardware

The Version UVH compatible canisters are authorized to store spent fuel assemblies with the minimum cooling time as a function of the assembly burnup.

The burnup and cooling time for every fuel assembly loaded into the MPC must satisfy the following equation:

$$Ct = A \cdot Bu^4 + B \cdot Bu^3 + C \cdot Bu^2 + D\ Bu + E$$

where,

Ct = Minimum cooling time (years),

Bu = Assembly-average burnup (MWd/mtU),

A, B, C, D, E = Polynomial coefficients listed in Tables 2.I.1.7, 2.I.1.8, and 2.I.1.9.

The coefficients for the above equation for the fuel assembly in an individual cell depend on the heat load limit in that cell. Tables 2.I.1.7, 2.I.1.8, and 2.I.1.9 list the coefficients for several heat load limit ranges for the Version UVH compatible canisters. Note that the heat load limits are only used for the lookup of the coefficients in that table, and do not imply any equivalency. Specifically, meeting the heat load limits established in Subsection 2.I.1.1 and Chapter 4.I is not a substitute for meeting burnup and cooling time limits herein, and vice versa.

Table 2.I.1.1

HI-STORM FW Version UVH Compatible MPCs

MPC Type	Fuel Storage Characteristics
MPC-37	See Section 2.1
MPC-44	See Section 2.1
MPC-89	See Section 2.1

TABLE 2.I.1.2 HI-STORM FW UVH MPC-37 HEAT LOAD DATA			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Total Heat Load (kW): 29			
Maximum Section Heat Load (kW): 3 (Note 1)			
Region No.	Decay Heat Limit per Cell, kW (Note 2)	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.784	9	7.054
2	1.568	12	18.816
3	1.568	16	25.088
Note 1: Figure 2.I.2.1 identifies the cell locations in each section. Note 2: Maximum total heat load, maximum 1/8th section heat load (as noted in Table 2.I.1.3) and specific cell heat load limits may need to be adjusted in accordance with Section 2.I.1.1. Note 3: This pattern can be modified to develop regionalized patterns in accordance with the requirements in Table 2.I.1.3.			

TABLE 2.I.1.3
HI-STORM FW UVH MPC-37 REQUIREMENTS ON DEVELOPING REGIONALIZED HEAT LOAD PATTERNS (See Figure 2.I.2.1)

1. Pattern-specific total heat load must be equal to 29 kW
2. Section Heat Load must be equal to 3.625 kW, calculated per Figure 2.I.2.1, and pattern must be 1/8th symmetric
3. Maximum Allowable Decay Heat per Cell in Region 1 is 0.784 kW
4. Maximum Allowable Decay Heat per Cell in Region 2 is 1.568 kW
5. Maximum Allowable Decay Heat per Cell in Region 3 is 1.568 kW
6. Pattern-specific Decay Heat in a storage cell may need to be adjusted to meet items 1 and 2
7. Pattern-specific decay heat for any storage cell in Region 1 may be determined by reducing the allowable in Region 1 of Table 2.I.1.2 by Δ and pattern-specific decay heat for any storage cell in Regions 2 and 3 may be determined by increasing the allowable in Region 2 and/or Region 3 of Table 2.I.1.2 by the same Δ.
8. Pattern-specific decay heat for any storage cell in Region 2 may be determined by reducing the allowable in Region 2 of Table 2.I.1.2 by θ and pattern-specific decay heat for any storage cell in Region 3 may be determined by increasing the allowable in Region 3 of Table 2.I.1.2 by the same θ. This θ may not be added to other cells in Region 2.
9. Items 1 through 8 need to be scaled in accordance with Section 2.I.1.1 for non-standard active fuel lengths.

General Note – The limits developed for the patterns are maximums, and any assembly with a heat load less than those limits can be loaded in the applicable cell, provided it meets all other CoC requirements.

TABLE 2.I.1.4 HI-STORM FW UVH MPC-89 HEAT LOAD DATA			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Total Heat Load (kW): 29			
Maximum Section Heat Load (kW): 3 (Note 1)			
Region No.	Decay Heat Limit per Cell, kW (Note 2)	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.326	9	2.932
2	0.652	40	26.08
3	0.652	40	26.08
Note 1: Figure 2.I.2.2 identifies the cell locations in each section. Note 2: Maximum total heat load, maximum 1/8th section heat load (as noted in Table 2.I.1.5) and and specific cell heat load limits may need to be adjusted in accordance with Section 2.I.1.1 Note 3: This pattern can be modified to develop regionalized patterns in accordance with the requirements in Table 2.I.1.5.			

TABLE 2.I.1.5
HI-STORM FW UVH MPC-89 HEAT LOAD DATA REQUIREMENTS ON DEVELOPING REGIONALIZED HEAT LOAD PATTERNS (See Figure 2.I.2.2)

1. Pattern-specific total heat load must be equal to 29 kW
2. Section Heat Load must be equal to 3.625 kW, calculated per Figure 2.I.2.2, and pattern must be 1/8th symmetric
3. Maximum Allowable Decay Heat per Cell in Region 1 is 0.326 kW
4. Maximum Allowable Decay Heat per Cell in Region 2 is 0.652 kW
5. Maximum Allowable Decay Heat per Cell in Region 3 is 0.652 kW
6. Pattern-specific Decay Heat in a storage cell may need to be adjusted to meet items 1 and 2
7. Pattern-specific decay heat for any storage cell in Region 1 may be determined by reducing the allowable in Region 1 of Table 2.I.1.4 by Δ and pattern-specific decay heat for any storage cell in Regions 2 and 3 may be determined by increasing the allowable in Region 2 and/or Region 3 of Table 2.I.1.4 by the same Δ.
8. Pattern-specific decay heat for any storage cell in Region 2 may be determined by reducing the allowable in Region 2 of Table 2.I.1.4 by θ and pattern-specific decay heat for any storage cell in Region 3 may be determined by increasing the allowable in Region 3 of Table 2.I.1.4 by the same θ. This θ may not be added to other cells in Region 2.
9. Items 1 through 8 need to be scaled in accordance with Section 2.I.1.1 for non-standard active fuel lengths.

General Note – The limits developed for the patterns are maximums, and any assembly with a heat load less than those limits can be loaded in the applicable cell, provided it meets all other CoC requirements.

TABLE 2.I.1.6 HI-STORM FW UVH MPC-44 HEAT LOAD DATA (See Figure 1.2.1d for Cell Identification)	
Number of Regions:	1
Number of Storage Cells:	44
Maximum Design Basis Heat Load (kW):	28
Decay Heat Limit per Cell, kW:	0.636

Table 2.I.1.7

Burnup and Cooling Time Fuel Qualification Requirements for MPC-37

Cell Decay Heat Load Limit (kW)	Polynomial Coefficients, see Subsection 2.I.1.2				
	A	B	C	D	E
≤ 0.784	-3.47307e-18	+6.86560e-13	-3.32645e-08	+6.67420e-04	-1.49033e+00
0.784 < decay heat ≤ 1.568	+2.60495e-19	-2.97225e-14	+1.63620e-09	+2.81518e-05	+1.01927e+00

Table 2.I.1.8

Burnup and Cooling Time Fuel Qualification Requirements for MPC-44

Cell Decay Heat Load Limit (kW)	Polynomial Coefficients, see Subsection 2.I.1.2				
	A	B	C	D	E
≤ 0.659	-1.04277e-17	+1.59059e-12	-6.70632e-08	+1.13293e-03	-3.01429e+00
0.659 < decay heat ≤ 1.318	+5.95983e-19	-4.84920e-14	+1.50900e-09	+5.37342e-05	+9.76479e-01

Table 2.I.1.9

Burnup and Cooling Time Fuel Qualification Requirements for MPC-89

Cell Decay Heat Load Limit (kW)	Polynomial Coefficients, see Subsection 2.I.1.2				
	A	B	C	D	E
≤ 0.326	-1.83052e-18	+4.29713e-13	-2.19962e-08	+4.76650e-04	-7.28771e-01
0.326 < decay heat ≤ 0.652	+5.33388e-19	-6.36638e-14	+2.76988e-09	+6.09709e-06	+9.75025e-01

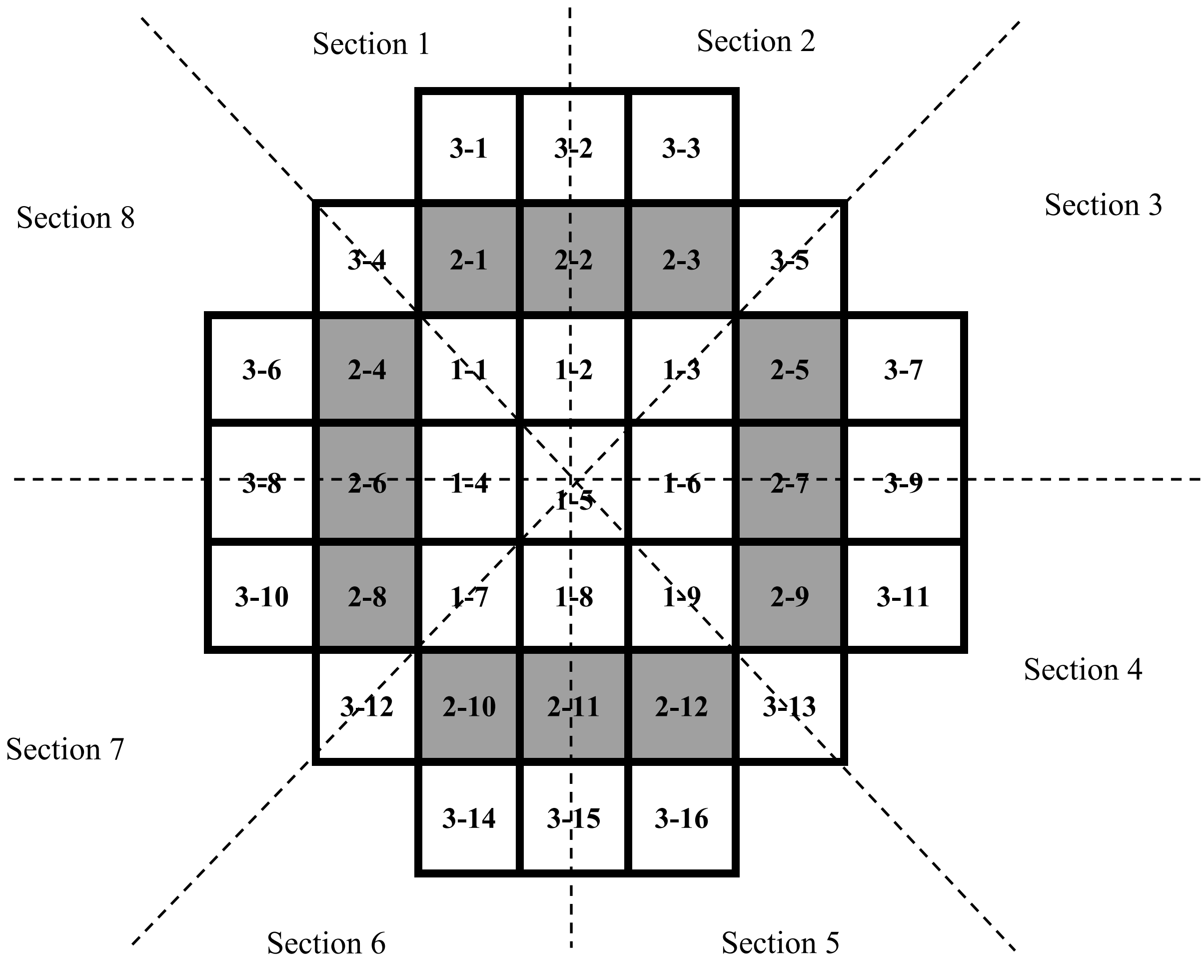

Figure 2.I.1.1 – MPC-37 Cell and Section Identification

To calculate the per section heat load, the following apply, where Q represents the heat load in the identified cell in kW.

Section 1: $Q_{3-1} + Q_{2-1} + ½Q_{3-2} + ½Q_{3-4} + ½Q_{2-2} + ½Q_{1-1} + ½Q_{1-2} + 1/8Q_{1-5}$
Section 2: $Q_{3-3} + Q_{2-3} + ½Q_{3-2} + ½Q_{3-5} + ½Q_{2-2} + ½Q_{1-3} + ½Q_{1-2} + 1/8Q_{1-5}$
Section 3: $Q_{2-5} + Q_{3-7} + ½Q_{1-6} + ½Q_{3-5} + ½Q_{2-7} + ½Q_{1-3} + ½Q_{3-9} + 1/8Q_{1-5}$
Section 4: $Q_{2-9} + Q_{3-11} + ½Q_{1-6} + ½Q_{1-9} + ½Q_{2-7} + ½Q_{3-13} + ½Q_{3-9} + 1/8Q_{1-5}$
Section 5: $Q_{2-12} + Q_{3-16} + ½Q_{1-8} + ½Q_{1-9} + ½Q_{2-11} + ½Q_{3-13} + ½Q_{3-15} + 1/8Q_{1-5}$
Section 6: $Q_{2-10} + Q_{3-14} + ½Q_{1-8} + ½Q_{1-7} + ½Q_{2-11} + ½Q_{3-12} + ½Q_{3-15} + 1/8Q_{1-5}$
Section 7: $Q_{2-8} + Q_{3-10} + ½Q_{1-4} + ½Q_{1-7} + ½Q_{2-6} + ½Q_{3-12} + ½Q_{3-8} + 1/8Q_{1-5}$
Section 8: $Q_{2-4} + Q_{3-6} + ½Q_{1-4} + ½Q_{1-1} + ½Q_{2-6} + ½Q_{3-4} + ½Q_{3-8} + 1/8Q_{1-5}$

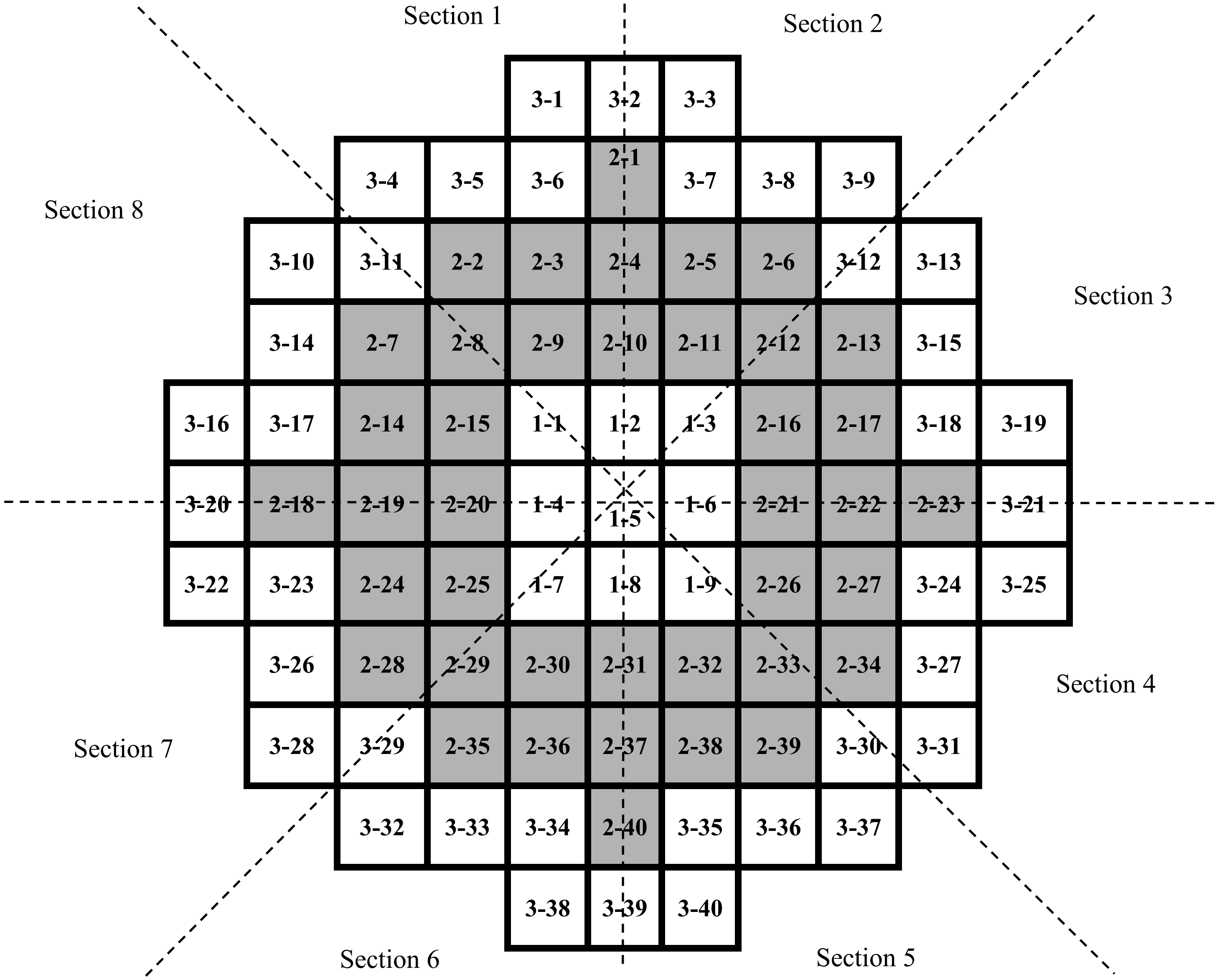

Figure 2.I.1.2 – MPC-89 Cell and Section Identification

To calculate the per section heat load, the following apply, where Q represents the heat load in the identified cell in kW.

Section 1: $Q_{3-1} + Q_{3-4} + Q_{3-5} + Q_{3-6} + Q_{2-2} + Q_{2-3} + Q_{2-9} + \frac{1}{2}Q_{3-2} + \frac{1}{2}Q_{2-1} + \frac{1}{2}Q_{2-4} + \frac{1}{2}Q_{2-10} + \frac{1}{2}Q_{1-2} + \frac{1}{2}Q_{1-1} + \frac{1}{2}Q_{2-8} + \frac{1}{2}Q_{2-11} + 1/8Q_{1-5}$

Section 2: $Q_{3-3} + Q_{3-7} + Q_{3-8} + Q_{3-9} + Q_{2-5} + Q_{2-6} + Q_{2-11} + \frac{1}{2}Q_{3-2} + \frac{1}{2}Q_{2-1} + \frac{1}{2}Q_{2-4} + \frac{1}{2}Q_{2-10} + \frac{1}{2}Q_{1-2} + \frac{1}{2}Q_{1-3} + \frac{1}{2}Q_{2-12} + \frac{1}{2}Q_{3-12} + 1/8Q_{1-5}$

Section 3: $Q_{3-13} + Q_{2-13} + Q_{3-15} + Q_{2-16} + Q_{2-17} + Q_{3-18} + Q_{3-19} + \frac{1}{2}Q_{1-6} + \frac{1}{2}Q_{2-21} + \frac{1}{2}Q_{2-22} + \frac{1}{2}Q_{2-23} + \frac{1}{2}Q_{3-21} + \frac{1}{2}Q_{1-3} + \frac{1}{2}Q_{2-12} + \frac{1}{2}Q_{3-12} + 1/8Q_{1-5}$

Section 4: $Q_{2-26} + Q_{2-27} + Q_{3-24} + Q_{3-25} + Q_{2-34} + Q_{3-27} + Q_{3-31} + \frac{1}{2}Q_{1-6} + \frac{1}{2}Q_{2-21} + \frac{1}{2}Q_{2-22} + \frac{1}{2}Q_{2-23} + \frac{1}{2}Q_{3-21} + \frac{1}{2}Q_{1-9} + \frac{1}{2}Q_{2-33} + \frac{1}{2}Q_{3-30} + 1/8Q_{1-5}$

Section 5: $Q_{2-32} + Q_{2-38} + Q_{2-39} + Q_{3-35} + Q_{2-36} + Q_{3-37} + Q_{3-40} + \frac{1}{2}Q_{1-8} + \frac{1}{2}Q_{2-31} + \frac{1}{2}Q_{2-37} + \frac{1}{2}Q_{2-40} + \frac{1}{2}Q_{3-39} + \frac{1}{2}Q_{1-9} + \frac{1}{2}Q_{2-33} + \frac{1}{2}Q_{3-30} + 1/8Q_{1-5}$

Section 6: $Q_{2-30} + Q_{2-35} + Q_{2-36} + Q_{3-32} + Q_{2-33} + Q_{3-34} + Q_{3-38} + \frac{1}{2}Q_{1-8} + \frac{1}{2}Q_{2-31} + \frac{1}{2}Q_{2-37} + \frac{1}{2}Q_{2-40} + \frac{1}{2}Q_{3-39} + \frac{1}{2}Q_{1-7} + \frac{1}{2}Q_{2-29} + \frac{1}{2}Q_{3-29} + 1/8Q_{1-5}$

Section 7: $Q_{2-25} + Q_{2-24} + Q_{3-23} + Q_{3-22} + Q_{2-28} + Q_{3-26} + Q_{3-28} + \frac{1}{2}Q_{1-4} + \frac{1}{2}Q_{2-20} + \frac{1}{2}Q_{2-19} + \frac{1}{2}Q_{2-18} + \frac{1}{2}Q_{3-20} + \frac{1}{2}Q_{1-7} + \frac{1}{2}Q_{2-29} + \frac{1}{2}Q_{3-29} + 1/8Q_{1-5}$

Section 8: $Q_{2-15} + Q_{2-14} + Q_{3-17} + Q_{3-16} + Q_{2-7} + Q_{3-14} + Q_{3-10} + \frac{1}{2}Q_{1-4} + \frac{1}{2}Q_{2-20} + \frac{1}{2}Q_{2-19} + \frac{1}{2}Q_{2-18} + \frac{1}{2}Q_{3-20} + \frac{1}{2}Q_{1-1} + \frac{1}{2}Q_{2-8} + \frac{1}{2}Q_{3-11} + 1/8Q_{1-5}$

2.I.2 Design Loadings:

2.I.2.1 Mechanical Loadings

The Design Basis Loads (DBLs) applicable to the Version UVH overpack are summarized in Table 2.I.2.1 wherein the justification for their admissibility is also provided obviating the need for a structural evaluation in Chapter 3.I. Additional justification is provided in Section 3.I.3.

a) Loadings unique to Version UVH by virtue of its vent-less design arise from the potential for the internal pressure in the cavity to fall below the initial fill pressure under extreme cold conditions. To bound all potential pressure variations, the Design Basis Internal Pressure (DBIP) in the cask cavity is set equal to *full vacuum* on the lower end and a bounding value on the upper end in Table 2.I.2.3.

b) Accident External Pressure: A state of external pressure may arise if the cask is submerged by flood waters or is exposed to pressure wave from an explosive device. The Accident External Pressure (AEP) for this condition is set down at a value which is based on the site-specific loadings being used at numerous operating ISFSIs (Table 2.I.2.3).

c) Accident Internal Pressure: The internal gas pressure in the Version UVH cask cavity may rise under hypothetical accident conditions. To envelope the potential pressure variations, the Accident Internal Pressure (AIP) in the cask cavity is set to a bounding value in Table 2.I.2.3.

d) Acceptance Criterion: It is necessary to demonstrate that the dual wall cask shell structure, cask's base plate and its closure lid can withstand all loadings without exceeding the stress limits set forth in Section III Subsection NF of the ASME Code.

2.I.2.2 Thermal Loadings

The Thermal Loadings applicable to the Version UVH overpack are summarized in Table 2.I.2.2, with considerations as follows. The ambient temperatures for the Version UVH overpack operations are specified in Table 2.I.2.4. Temperature limits in Table 2.2.3 remain applicable for the Version UVH system.

a. Normal Condition of Storage (T-1): The reference ambient conditions corresponding to the normal, off-normal, and accident conditions of storage are provided in Table 2.I.2.2. These environmental conditions have been determined to bound their respective meteorological data for the entire continental United States. Unsheltered condition of storage requires inclusion of insolation to the storage system as external heat input.

b. Accident Condition of Storage: This condition is characterized by an elevated ambient temperature (Table 2.I.2.2).

c. Design Basis Fire: The accident condition design temperature limits for the Version UVH system are specified in Table 2.2.3. The specified fuel cladding temperature limits are based on the temperature limits specified in ISG-11, Rev 3.

Table 2.I.2.1; Evaluation of the Design Basis Loadings for the Version UVH Storage Cask			
Applicable Loading Case from Tables 2.2.6, 2.2.7 and 2.2.13	**Load Case Description**	**Subsection in the main report where the loading is explained**	**Safety Consideration and Conclusion**
AD	Moving Floodwaters Moving Floodwater with loaded HI-STORM on the pad	2.2.3	Determine the constant flood velocity that will not lead to sliding or overturning the overpack. Because the weight of the loaded UVH cask is greater than the benchmark overpack (HI-STORM FW Version E) due to the removal of the vent openings and the use of high density concrete, the resistance to sliding and overturning is greater even though the UVH cask outer diameter (OD) is smaller. This is demonstrated in Subsection 3.I.3.4. Therefore, the admissible flood water velocity in Subparagraph 3.4.4.1.1 is conservative.
AE	Design Basis Earthquake (DBE) Loaded HI-STORMs arrayed on the ISFSI pad subject to ISFSI's DBE	2.2.3	This case involves determining the maximum magnitude of the earthquake that meets the acceptance criteria of Paragraph 2.2.3(g). This is presented in Subsection 3.I.3.6. The discussion in Subparagraph 3.4.4.1.2 is applicable to Version UVH cask.
AC	Strike by a Tornado-borne Missile A large, medium or small tornado missile strikes a loaded HI-STORM on the ISFSI pad or a loaded HI-TRAC	2.2.3	This criterion requires that the acceptance criteria of 2.2.3(e) be met. Because the weight of the loaded UVH cask is greater than the benchmark overpack (HI-STORM FW Version E) due to the removal of the vent openings and the use of high density concrete, the resistance to sliding and overturning is greater even though the UVH cask outer diameter (OD) is smaller. This is demonstrated in Subsection 3.I.3.7. The absence of vent openings is a positive structural advantage against missile penetrations for Version UVH.
AH	Design Basis External Pressure Loaded HI-STORMs arrayed on the ISFSI pad subject to destabilizing external pressure from blast/explosion	3.1.2.1	Determine the constant de-stabilizing external pressure acting on the overpack that will not result in its sliding or overturning. Because the weight of the loaded UVH cask is greater than the standard overpack due to the removal of the vent openings and the use of high density concrete, the resistance to sliding and overturning is greater even though the UVH cask outer diameter (OD) is smaller. The evaluation is presented in Subsection 3.I.3.5.
AA	Non-Mechanistic Tip-Over A loaded HI-STORM is assumed to tip over and strike the pad.	2.2.3	Version UVH's response to the tip-over event will be different to the cask designs analyzed in Chapter 3 to the acceptance criteria of Paragraph 2.2.3(b) because of the change in MPC-to-cask body gap, elimination of MPC guide tubes and a different lid design. Therefore, new tip-over analyses are performed in

Table 2.I.2.1; Evaluation of the Design Basis Loadings for the Version UVH Storage Cask			
Applicable Loading Case from Tables 2.2.6, 2.2.7 and 2.2.13	**Load Case Description**	**Subsection in the main report where the loading is explained**	**Safety Consideration and Conclusion**
			Subsection 3.I.3.8 for the Version UVH cask design with MPC-37, MPC-44 and MPC-89.
HC	Handling of Cask	2.2.3	The methodology for evaluating the handling loads in Section 3.4.3 remains applicable. The lifting analysis of Version E cask using bounding lifted weight in Table 3.2.8 remains applicable for Version UVH cask. The combination of lifted load and internal pressure in UVH cask is considered in Subsection 3.I.3.2. Site specific verification of the handling loads is required under the plant's §72.212 mandated by the system's CoC.
NA	Snow Load	2.2.1	The Design Basis snow load in Chapter 3 is used to evaluate the Version UVH Closure Lid in Chapter 3.I.

Table 2.I.2.2; Governing Thermal Loading Conditions

Thermal Loading ID	Caption of Loading	Applicable FSAR paragraph	Comments
T-1	Normal Condition of Storage	2.2.1	The ISG-11, Rev 3 peak cladding temperature limits specified in Table 2.2.3 of this FSAR must be met. In addition, the temperature of proximate safety significant materials must meet applicable temperature limits as described in Table 2.2.3.
T-2	Design Basis Fire	2.2.3.3	
T-3	Extreme Environment Temperatures	2.2.3.14	
T-4	Burial under debris	2.2.3.12	The Version UVH overpack does not experience a significant loss of its heat dissipation capability under this event due to its lack of ventilation passages and lower allowable heat loads compared to the standard overpack design. Therefore, the evaluation for the standard overpack design remains conservative.

Table 2.I.2.3; Pressure Loadings for the Version UVH Canister Storage Cask		
Loading	**Value, psig**	**Comment**
Design Basis Minimum Internal Pressure	-14.7	Corresponds to full vacuum
Design Basis Maximum Internal Pressure	10	Bounding internal pressure under normal and off-normal conditions.
Accident External Pressure	60	Bounding steady state pressure assumed to act on all external surfaces of the overpack
Accident Internal Pressure	15	Bounding internal pressure under hypothetical conditions

Table 2.I.2.4; ENVIRONMENTAL TEMPERATURES	
HI-STORM FW Version UVH Overpack	
Condition	**Temperature (ºF)**
Normal Ambient Temperature	70
Soil Temperature	70
Off-Normal Ambient Temperature	See Table 2.2.2
Extreme Ambient Temperature	See Table 2.2.2
Short-Term Operations	See Table 2.2.2

2.I.3 Safety Protection Systems

There is no change in the safety protection systems described in Section 2.3 required by the introduction of the Version UVH overpack, except as noted below.

2.I.3.1 Cask Cooling

Unlike the ventilated overpack design described in the main body of this FSAR, the Version UVH overpack does not rely on ventilation passages for its means of cooling. Heat dissipation from the MPC in the Version UVH overpack is primarily facilitated through conduction and radiation based heat transfer. Heat is then carried to the overpack external surfaces through conduction and is then rejected to the environment through natural convection and radiation.

2.I.4 Decommissioning Considerations

The decommissioning considerations described in Section 2.4 remain applicable.

2.I.5 Safety Conclusions

The evaluations in this supplement show that:

- The loadings specified in Chapter 2 for the classical HI-STORM FW ventilated overpacks that are also applicable to the unvented Version UVH overpack are satisfied by it.

- Additional loadings - internal and external pressures have been identified for Version UVH overpack that warrant analysis to demonstrate safety compliance with the acceptance criteria in this supplement.

- Version UVH Closure Lid requires analyses to demonstrate safety compliance with the acceptance criteria in main FSAR.

- The non-mechanistic tip-over analysis of the freestanding Version UVH system needs to be performed to demonstrate the satisfaction of acceptance criteria in Paragraph 2.2.3(b).

- A thermal analysis of the Version UVH is warranted to ensure peak cladding temperature remains below ISG-11 limits.

- All other components of the Storage system are unaffected by the choice of the version of the overpack employed in the Storage system.

CHAPTER 3.I: STRUCTURAL EVALUATION

3.I.0 Overview

This chapter contains the structural safety analysis of the HI-STORM FW storage system containing the Version UVH overpack (hereafter referred to as the *Storage System* for brevity) illustrated in Figure 1.I.1.1 and in the Licensing drawing package in Section 1.I.5. The structural evaluation for Version UVH under all applicable loadings, including pressure and the non-mechanistic tip-over event, are discussed in Chapter 2.I and evaluated in Section 3.I.3. The loading scenarios for Version UVH overpack that are bounded by the existing analyses in Chapter 3 of this FSAR are presented in Table 2.I.2.1 and Section 3.I.3.

3.I.1 Structural Design

The design information provided in Section 3.1 remains applicable except that the storage cask has no inlet or outlet vents and the space between the cask cavity and the MPC is confined. To ensure a sequestered space outside the MPC, the interface between the closure lid and the cask body is equipped with a weather-resistant gasket.

The absence of vents confers additional structural resistance to Version UVH to certain mechanical loadings such as the Design Basis penetrant missiles considered in Subsection 3.4.4 of this report. Indeed, as the evaluation narrative in Table 2.I.2.1 and Section 3.I.3 demonstrates, from the structural standpoint, the Version UVH is either similar or better in terms of the safety margins established for the system components in the main FSAR except for the cask internal pressure and the non-mechanistic tip-over evaluations which are distinct. All evaluations to demonstrate structural integrity of various components are discussed in Section 3.I.3.

As this chapter envisages no change to the MPCs or their contents or to the HI-TRAC transfer casks, all safety information on them in Chapter 3 remains fully applicable. The only new calculations for MPC are limited to ensuring that the temperature field in the enclosure vessel is bounded by those used in the Subsection 3.4.3 evaluations under normal and off-normal conditions. The safety evaluation of the Design Basis Loadings (DBLs) for Version UVH overpack is limited to ensuring that the overpack's response remains acceptable under the design criteria and features unique to Version UVH, which, as stated in Section 2.I.2, consists of Design Basis internal pressure loadings and cask stability evaluations. In addition, since the Version UVH overpack lid is different from those designs evaluated in Subsections 3.4.3 and 3.4.4, a separate lid evaluation is performed under all applicable loadings as described in Section 3.I.3.

3.I.2 Structural Model

The cask body is simulated as dual shell structure with discrete radial connectors, the shielding concrete serves no structural function except to keep the two shells from deflecting laterally into the space occupied by the concrete. The baseplate of the cask is modeled as a flat plate buttressed by radial ribs and held from lateral deflection at its connection with the edges of the two shells.

The cask lid is secured to the body using four large anchor bolts similar to the standard FW design, and it has shielding concrete to keep the top and bottom plates and outer shell from deflecting into the space occupied by the concrete. In addition, there are diagonal stiffener plates through the thickness of the lid to reinforce the structure and to provide interfacing lift points for handling of the lid.

3.I.3 Safety Analyses

As discussed in Chapter 2.I and Section 3.I.1, multiple evaluations are performed to demonstrate structural integrity of HI-STORM FW Version UVH Storage System, namely,

a) Evaluation of MPC containment boundary under normal and off-normal conditions using pressure limits from Table 2.2.1 and temperature profiles from thermal analyses supporting Chapter 4.I.

b) Evaluation of Version UVH cask under internal and external pressure loads.

c) Evaluation of Version UVH closure lid under lifting and snow load conditions.

d) Evaluation of Version UVH cask's stability in the event of flood.

e) Evaluation of Version UVH cask's stability in the event of explosion or blast.

f) Evaluation of Version UVH cask's stability in the event of earthquake.

g) Evaluation of Version UVH cask in the event of tornado (wind and missile impacts).

h) Evaluation of Version UVH cask in the event of non-mechanistic tip-over.

3.I.3.1 MPC Containment Boundary Evaluation

Using the ANSYS finite element model of MPC described in Paragraph 3.4.3.2, two separate analyses are added to demonstrate structural integrity in Supplement 1 of [3.4.13] as described below.

i) The long-term normal condition internal pressure limit in Table 2.2.1 is applied to MPC lid, shell and baseplate along with temperature contour obtained from normal condition thermal analysis in Chapter 4.I. The primary and secondary stresses in MPC lid, shell and baseplate are then compared against ASME NB Level A stress limits obtained at bounding component temperatures. It is demonstrated in Supplement 1 of [3.4.13] that all safety factors are greater than 1.0.

ii) The off-normal condition internal pressure limit in Table 2.2.1 is applied to MPC lid, shell and baseplate along with temperature contour obtained from off-normal condition thermal analysis in Chapter 4.I. The primary and secondary stresses in MPC lid, shell and baseplate are then compared against ASME NB Level B stress limits obtained at bounding component temperatures. It is demonstrated in Supplement 1 of [3.4.13] that all safety factors are greater than 1.0.

3.I.3.2 HI-STORM FW Version UVH Cask Pressure Loading

A 3-D finite element model of the HI-STORM FW Version UVH overpack is constructed in ANSYS [3.4.26] as shown in Supplement 43 of [3.4.13]. All plate and shell components are modeled using ANSYS solid elements (SOLID185 with at least three elements through thickness) and concrete is modeled using ANSYS solid element (SOLID65).

Six pressure loading cases (four normal, including lifting, and two accident) are evaluated in Supplement 43 of [3.4.13] to envelope all design basis internal and external pressure loadings in Table 2.I.2.2.

The primary membrane and membrane plus bending stresses in Version UVH overpack shells, base plate and lid are compared against ASME NF Level A (under normal and off-normal (conservative) pressure loadings) and Level D (under accident pressure loadings) stress limits obtained at bounding temperatures. It is demonstrated in Supplement 43 of [3.4.13] that all safety factors are greater than 1.0. In addition, it is demonstrated that outer shell of overpack does not collapse or buckle under the accident external pressure loading.

3.I.3.3 HI-STORM FW Version UVH Cask Closure Lid

The Version UVH closure lid is evaluated under the following load conditions in Supplement 46 of [3.4.13] using the same methodology and acceptance criteria used to evaluate standard, XL, domed and Version E closure lids in Subsections 3.4.3 and 3.4.4.

i) Lid lifting: It is demonstrated in Supplement 46 of [3.4.13] that the stresses in lid lifting points are less than NUREG-0612 and Regulatory Guide 3.61 stress limits obtained at bounding temperatures for the heaviest lid (bounded by maximum lid weight in Table 3.2.5). Also, the primary stresses in the remainder of lid structure, including welds, are shown to be less than ASME Code Subsection NF Level A stress limits obtained at bounding temperature.

ii) Snow load: It is demonstrated in Supplement 46 of [3.4.13] that under a bounding snow load, applied as pressure on top surface of closure lid, all primary stresses in the lid structure are less than ASME Code Subsection NF Level A stress limits obtained at bounding temperature.

3.I.3.4 HI-STORM FW Version UVH Cask Stability During Flood

The Version UVH cask's stability in the event of a flood is evaluated in Supplement 45 of [3.4.13] using Version UVH cask's dimensions and corresponding minimum weight following the guidance in Subparagraph 3.4.4.1.1 for standard FW cask. The analysis demonstrates that the maximum flood water velocity established in Subparagraph 3.4.4.1.1 remains governing for Version UVH cask.

3.I.3.5 HI-STORM FW Version UVH Cask Stability During Explosion

The Version UVH cask's stability in the event of an explosion is evaluated in Supplement 44 of [3.4.13] following the guidance in Subparagraph 3.1.2.1.d using Version UVH cask's dimensions and corresponding minimum weight. The analysis in Supplement 44 of [3.4.13] establishes the maximum static pressure that the cask can withstand without sliding or rocking (i.e., no incipient loss of kinematic stability). In case the blast pressures are greater at a Plant, a time-history based site-specific analysis using the applicable pressure-time pulse will be required to demonstrate that the cask will not slide excessively or overturn.

3.I.3.6 HI-STORM FW Version UVH Cask Stability During Earthquake

The discussion in Subparagraph 3.4.4.1.2 remains applicable to Version UVH cask. The combination of vertical and horizontal ZPAs of the earthquake that would cause Version UVH cask's incipient loss of kinematic stability are derived using static inequalities defined in Paragraph 2.2.3(g) in Supplement 47 of [3.4.13] using cask's dimensions and corresponding minimum weight. For earthquakes stronger than that defined by the inequalities in Subsection 2.2.3(g), it is necessary to perform a dynamic analysis per Subparagraph 3.4.4.1.2.

3.I.3.7 HI-STORM FW Version UVH Cask During Tornado

The Version UVH cask's stability in the event of a tornado (wind and missiles) is evaluated following the guidance in Subparagraph 3.4.4.1.3 for standard FW cask in Appendix I of [3.4.15] using Version UVH cask's dimensions and corresponding minimum weight. The analysis demonstrates that the maximum cask displacements due to sliding and rocking are governed by the results for standard FW cask.

The penetration analysis for Version UVH cask using the wind and missile characteristics defined in Tables 2.2.4 and 2.2.5 is presented in Appendix J of [3.4.15] demonstrating all results are acceptable.

3.I.3.8 Non-Mechanistic Tip-over

Non-mechanistic tip-over of the freestanding HI-STORM FW system consisting of the Version UVH overpack and three variants of MPC (MPC-37, MPC-44 and MPC-89) is considered herein. The solution uses the same methodology that was employed in the system's original certification documented in Subparagraph 3.4.4.1.4. The physical problem subject to the present analysis is different from the original problem in two respects; they are:

(a) As ascertained in Chapter 4.I, there is smaller clearance between the MPC and the overpack under the design basis heat load and as a result, there are no MPC guide tubes that participate in the cask's dynamics during its impact with pad.

(b) The top lid-to-cask body connectivity has been improved such that the lid strikes the ISFSI pad without applying any shear load on the anchor bolts. Thus, the impact of the lid is decoupled from that of the cask body which materially reduces the angular momentum of the cask as it collides with the pad during tip-over. The anchor bolts still serve the safety function of keeping the MPC confined within the cask's radiation shield against the centrifugal force generated by the tip-over event.

The LS-DYNA model of the system, therefore, considers the cask body and its MPC with a small annular clearance between them striking the pad as an assemblage with limited lateral kinematic freedom. The ability of the closure lid to constrain the MPC within the confines of the overpack is also evaluated. The target foundation properties per Tables 2.2.9 and 2.I.0.1 are utilized. In case the target properties are not bounded by those in Table 2.2.9 and 2.I.0.1, a site-specific analysis using the model described in [3.4.31] will be required to demonstrate satisfaction of acceptance criteria in Paragraph 2.2.3(b).

The details of the finite element model, input data and results are archived in the calculation package [3.4.31]. The following conclusions demonstrate that all safety criteria are satisfied for the Version UVH cask with MPC-37, MPC-44 and MPC-89 basket designs.

i. The lateral deflection of the most heavily loaded basket panel in the active fuel region complies with the deflection criterion in Table 2.2.11.
ii. The shims in MPC-44 basket remain attached to it maintaining its physical integrity.
iii. The plastic strains in the MPC enclosure vessel remain below the allowable material plastic strain limit.
iv. The cask closure lid does not dislodge after the tipover event, i.e., the closure lid bolts remain in-tact.
v. The lid or the cask body do not suffer any gross loss of shielding.

Table 3.I.3.1: ON-ISFSI WEIGHTS OF LOADED HI-STORM FW VERSION UVH

Scenario	Weight (kilo-pounds)
Shortest cask loaded with minimum fuel	283
Tallest cask fully loaded	450

3.I.4 Safety Conclusions

The structural evaluation of the Version UVH storage cask under the loading conditions unique to it, described in Section 3.I.3, demonstrates that the stresses in all cask components, namely the base plate, the dual shell structure and the closure lid weldments are below the ASME Code limits with significant margins. The structural analysis of the HI-STORM Version UVH closure lid demonstrates that the stresses in lid components are below the stress limits in ASME Code under all loading conditions. In addition, the MPC confinement boundary continues to satisfy the established acceptance criteria under temperature profiles unique to Version UVH Storage System under all loading conditions.

The stability evaluations of the Version UVH storage cask under tornado wind and missile impacts, explosion, earthquake and flood are also performed.

The structural integrity of the Version UVH cask system (with MPC-37, MPC-44 and MPC-89 baskets) is also demonstrated in the event of a non-mechanistic tip-over.

The structural safety of the Storage System under all other loadings germane to the ventilated cask model treated in the main body of this FSAR is established in Table 2.I.2.1. Therefore, the Version UVH overpack is proven to meet all structural criteria applicable to the HI-STORM FW Canister Storage System in this FSAR.

CHAPTER 4.I: THERMAL EVALUATION

4.I.0 Overview

In this supplement to Chapter 4, the thermal compliance of the HI-STORM FW system containing the Version UVH overpack to the ISG-11 Rev 3 [4.1.4] and other limits specified in Chapter 2 is considered. In particular, the thermal acceptance criteria provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF) and other Confinement Boundary components in the MPC, and on the maximum permissible pressure in the MPC confinement space under certain operating scenarios. Specifically, the requirements are:

i. The fuel cladding temperature must meet the temperature limit under normal, off-normal, and accident conditions appropriate to its burnup level and condition of storage or handling set forth in Table 4.3.1[1].

ii. The maximum internal pressure of the MPC and the air annulus should remain within their design pressures for normal, off-normal, and accident conditions set forth in Table 2.2.1.

iii. The temperatures of the cask materials shall remain below their allowable limits set forth in Table 2.2.3 under all scenarios.

As discussed in Section 1.I.1, Version UVH is characterized by a gasketed storage cavity space which encloses the MPC. Therefore, the sole path for the rejection of the spent fuel decay heat to the ambient air is by convection and radiation from the cask's external surface (i.e., no assist from ventilation of the MPC unlike the standard HI-STORM FW system). The heat from the canister surfaces is delivered to the overpack's inside surface primarily by radiation and conduction with convection playing a negligible role. The heat deposited on the overpack's inside surface is conducted to the outside through conduction along the inter-shell ribs and through the shielding concrete. [PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

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The safety evaluation of the Storage system is, therefore, carried out through a detailed 3-D Computational Fluid Dynamics (CFD) analysis of the Storage system on the QA validated Code, Fluent (which has been used in all thermal safety analyses in all Holtec dockets, including all analyses documented in Chapter 4 of this FSAR), using a set of conservative assumptions that seek to overstate the computed temperatures, summarized in this supplement.

[1] All table references without a Roman numeral in the second place indicate that they are in the main report (i.e., not in a supplement)

4.I.1 Discussion

The MPC is completely surrounded by the overpack, and the transmission of MPC's decay heat to the overpack occurs mainly by a combination of conduction, convection and radiation across the small annulus gap. The outward transmission of heat across the Version UVH body is facilitated by the conduction through the high-density concrete and through the radial connectors. Therefore, the conductivity of the overpack's concrete and the number and thickness of the radial connectors are important parameters for thermal analyses of the system.

4.I.1.1 Allowable Heat Load Patterns

MPC-44 is licensed only for uniform heat load pattern as defined in Table 4.I.1.1. The discussion of allowable heat load patterns for MPC-37 and MPC-89 is presented below.

The selection of the location in the basket where a fuel assembly of a specific fuel heat load has the most pronounced effect on the peak fuel cladding temperature. Because individual fuel batches for fuel loading have different composition of specific heat loads, it is necessary to provide flexibility in the heat load pattern such that one CoC covers as many batches as possible. To that end, the approach of multi-region storage is generalized using the following strategy based on heuristic reasoning coupled with bounding pattern evaluations.

The definitions of the storage regions are the same as those described in Section 1.2 of the main SAR. The uniform heat load pattern, i.e. the maximum allowable decay heat is the same (say, q) for all the n cells, is designated as Pattern-0. Thus, the aggregate heat load Q_0 is given by

$$Q_0 = \mathrm{n.q}$$

Q_0 is defined in Table 4.I.1.1.

Any site-specific regionalized heat load pattern is subject to the following constraints:

1. The total heat load should be equal to Q_0.
2. The heat load pattern should exhibit 1/8th symmetry as defined in Figures 1.I.2-1 (MPC-37) and 1.I.2-2 (MPC-89). Thus, the total decay heat in any of the eight sections is equal to $Q_0/8$.
3. The maximum allowable decay heat per cell in the region r is limited by the following expression:

$$q_r = q \; for \; r = 1$$

$$= 2\mathrm{q} \; for \; r > 1$$

4. The minimum allowable decay heat per cell in regions 1 and 2 is 0 and that for region 3 is q.
5. Pattern-specific heat load for storage cells in Region 1 may be determined by reducing the allowable in Region 1 of Pattern-0 by Δ and heat load for any storage cell in Region 2 and 3 may be determined by increasing the allowable in Region 2 and/or Region 3 of Pattern-0 by the same Δ.

6. Pattern-specific heat load for storage cells in Region 2 may be determined by reducing the allowable in Region 2 of Pattern-0 by θ and heat load for any storage cell in Region 3 may be determined by increasing the allowable in Region 3 of Pattern-0 by the same θ.

The validity of the above generalized multi-region storage strategy within the confines of the above rules is demonstrated by parametric analyses for the bounding heat load patterns. Details of the methodology to identify these bounding heat load patterns are given in Section 4.I.1.3.

4.I.1.2 Fuel-length Dependent Allowance of Heat Loads

All the analyses identified in the above manner are performed for standard length PWR and BWR commercial spent fuel (CSF). The maximum allowable heat load per storage cell determined using the aforementioned process, is therefore, applicable for CSF with standard and longer active length fuels.

For fuel with shorter active fuel lengths than the standard active length defined in Table 4.I.1.2, the maximum storage cell-specific allowable heat loads are determined using the following scaling formula:

$$q_i = \frac{L}{L_{std}} * q_0, \qquad \text{where,}$$

q_i is maximum allowable cell-wise heat load,

L is the active length of the fuel,

L_{std} is the active length of the standard-length fuel used in the corresponding analysis for PWR or BWR in this supplement listed in Table 4.I.1.2, and

q_0 is maximum allowable heat load for the corresponding cell in the standard-length analysis based on the rules prescribed in Section 4.I.1.1.

Similarly, for fuel with a longer active fuel length than the standard active length defined in Table 4.I.1.2, the maximum storage cell-specific allowable heat loads are determined using the following scaling formula:

$$q_i = \sqrt{\frac{L}{L_{std}}} * q_0$$

4.I.1.3 Identification of Heat Load Patterns for Parametric Analyses[1]

To demonstrate the validity of the generalized multi-region storage strategy described in the preceding section, the following procedure is used to identify the bounding patterns:

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[1] As stated before, MPC-44 is licensed only for uniform heat load pattern, and therefore, this discussion is not applicable.

Therefore, following this strategy, several regionalized patterns for MPC-37 and MPC-89 are identified and evaluated using 3-D Computational Fluid Dynamics (CFD) models (Section 4.I.4.2).

4.I.1.4 Backfill Pressure Limits

The minimum and maximum initial helium backfill pressures for MPCs stored in Version UVH system are listed in Table 4.I.1.3. The air annulus between the MPC and the Version UVH overpack is backfilled such that the annulus air design pressure as specified in Table 4.I.1.3 is satisfied.

TABLE 4.I.1.1 TOTAL MAXIMUM ALLOWABLE HEAT LOADS FOR PWR AND BWR FUELS	
MPC Type (Fuel Type)	Maximum Allowable Heat Load
MPC-37 (PWR)	29 kW
MPC-44 (PWR)	28 kW
MPC-89 (BWR)	29 kW

TABLE 4.I.1.2 STANDARD ACTIVE FUEL LENGTHS USED IN THE THERMAL ANALYSES FOR PWR AND BWR FUELS	
MPC Type (Fuel Type)	Active Fuel Length (in)
MPC-37 (PWR)	144
MPC-44 (PWR)	144
MPC-89 (BWR)	150

Table 4.I.1.3
INITIAL BACKFILL PRESSURE LIMITS FOR MPC HELIUM AND ANNULUS AIR PRESSURE

Condition	MPC Helium Backfill Pressure Limits (psig)	Annulus Air Design Pressure (psig)
MPC-37	42.0 – 45.5 @70ºF	10 psig
MPC-44	41.0 – 44.0 @70ºF	10 psig
MPC-89	42.5 – 46.5 @70ºF	10 psig

4.I.2 Thermal Properties of Materials

The material property data in Section 4.2 applies except the conductivity of high-density concrete used in the cask body and lid is provided in Appendix 1.D to the HI-STORM 100 FSAR [4.I.2]. (Appendix 1.D has been the standard citation source for all HI-STORM models).

4.I.3 Specifications for Components

All applicable material temperature limits in Section 4.3 of the FSAR continue to apply to the HI-STORM FW Version UVH system.

4.I.4 Thermal Model and Evaluation of Normal Conditions of Storage

4.I.4.1 Thermal Model

The Storage system consists of the MPC standing upright on the cask's baseplate and the surrounding cask made of steel and plain concrete. The MPC thermal model is identical to that described in Section 4.4.1 of the main SAR. Since the analyses in this supplement employ standard active fuel length, accordingly, effective fuel properties of standard length PWR and BWR fuels are used. Taking advantage of the symmetry in MPC fuel loading pattern resulting from the patterns identified in Section 4.I.1.3, a quarter symmetric model of both the MPC and the overpack is employed.

The following methodology is employed to render the computed peak cladding temperature into an upper-bound of the value that would obtain in practice:

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4.I.4.2 Limiting MPC Configurations

Screening calculations are performed for various patterns identified in [4.I.1] for MPC-37 and MPC-89. These patterns are derived following the rules specified in Section 4.I.1.1. The peak fuel cladding temperatures for uniform and regionalized patterns for MPC-37, MPC-44 (uniform only), and MPC-89 (stand-alone casks, without accounting for the impact of neighboring casks) are summarized in Table 4.I.4.3. The most limiting MPC is adopted for all subsequent evaluations.

4.I.4.3 Test Model

The rationale for not requiring an experimental test model provided in Section 4.3 remains applicable in its entirety.

4.I.4.4 Normal Condition of Storage

The steady state thermal analysis to determine compliance with the temperature limits corresponding to the normal condition of storage consists of several discrete analyses, namely:

i. Storage system containing Version UVH and standard length MPC-37 (PWR canister)

ii. Storage system containing Version UVH and standard length MPC-89 (BWR canister).

iii. Storage system containing Version UVH and MPC-44 CBS assembly (PWR canister).

iv. Parametric analysis to demonstrate the validity of the Generalized Multi-Region (GMR) storage model explained in Section 4.I.1.3.

4.I.4.5 Impact of Neighboring Casks

As described in Section 4.4.2 of the main SAR, heat dissipation through the Version UVH overpack that is placed in an array is somewhat disadvantaged. The impact of the neighboring casks on the temperatures is particularly exacerbated in the case of Version UVH overpack compared to the standard HI-STORM FW system because of its high dependence on radiative heat transfer into the ambient due to lack of ventilation of the MPC. Therefore, to determine the impact of cask proximity, an example evaluation is performed for a 2xN layout shown in Figure 4.I.4.1. The methodology for this analysis is largely same as that used in Section 4.I.4.1 with the following differences:

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The computed temperatures and pressures are presented in Table 4.I.4.1 and meet the respective limits specified in this FSAR.

The above methodology can be adopted for any site-specific arrangement.

4.I.4.6 Results & Safety Conclusions

The results from the most-bounding configuration, determined in Section 4.I.4.2, are presented in Table 4.I.4.1. This evaluation includes the impact of the neighboring casks. The MPC cavity and annulus pressures are summarized in Table 4.I.4.2. It can be seen from the results that under the licensing-basis heat load:

i. The storage system containing the Version UVH satisfies the ISG-11 Rev. 3 fuel cladding and other temperature limits set down in this FSAR.

ii. The pressures inside the overpack and the MPC are below their respective design-basis limits (Chapter 2).

Table 4.I.4.1
SUMMARY OF COMPONENT TEMPERATURES UNDER NORMAL STORAGE CONDITIONS FOR THE MOST BOUNDING EVALUATION
(INCLUDING THE IMPACT OF NEIGHBORING CASKS)

Component	**Temperature, °C (°F)**
Fuel Cladding	363 (685)
MPC Basket	341 (646)
Shims/Basket Support Plates	272 (522)
MPC Shell	252 (486)
MPC Baseplate Note 1	222 (432)
MPC Lid Note 1	256 (493)
Overpack Inner Shell	220 (428)
Overpack Outer Shell	108 (226)
Overpack Body Concrete Note 2	138 (280)
Overpack Lid Concrete Note 2	111 (232)

Note 1: Maximum thru-thickness section average temperature is reported for these components.
Note 2: Maximum cross-sectional average temperature is reported for these components.

Table 4.I.4.2
SUMMARY OF MPC CAVITY AND HI-STORM UVH ANNULUS PRESSURES UNDER NORMAL STORAGE CONDITIONS FOR THE MOST BOUNDING CONFIGURATION Note 1

Condition	MPC-37 (psig)
MPC - Normal (Intact rods) - 1% rods rupture	 97.0 98.1
HI-STORM FW UVH Annulus	8.84

Note 1: 10% and 100% rod ruptures lead to release of significant amount of fission gases into the cavity, leading to an increase in the thermal conductivity of the cavity space. Since the MPC internal pressures under normal storage conditions for MPC-37, MPC-44, and MPC-89 in Version UVH [4.I.1] are bounded by those for MPC-37 in standard HI-STORM FW as reported in the main chapter, MPC pressure under off-normal and accident conditions will also be bounded by that for MPC-37 in standard HI-STORM FW.

Table 4.I.4.3
SUMMARY OF PEAK FUEL CLADDING TEMPERATURES FOR STAND-ALONE VERSION UVH CASKS

MPC TYPE & HEAT LOAD PATTERN	**TEMPERATURE, ^oC (^oF)**
MPC-37 - UNIFORM - REGIONALIZED	 353 (667) 356 (673) *
MPC-89 - UNIFORM - REGIONALIZED	 331 (628) 335 (635)
MPC-44, UNIFORM	348 (658)

*This MPC and the corresponding heat load pattern are implemented for all the licensing basis calculations.

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Figure 4.I.4.1: Schematic of a 2xN array of HI-STORM FW Version UVH Casks

4.I.5 Thermal Evaluation for Short Term Operations

As part of short-term operations, the HI-TRAC transfer cask is used, which remains unchanged. Since the maximum heat loads qualified for MPC-37, MPC-44, and MPC-89 in Version UVH system are significantly lower than those qualified for use in the standard HI-STORM FW version, the short-term operations' acceptance criteria will be satisfied with robust margins. Thus, the thermal evaluations presented in Section 4.5 of the main chapter remain bounding.

4.I.6 Off-Normal and Accident Events

4.I.6.1 Off-Normal Conditions

The most bounding fuel temperatures for Version UVH with MPC-37, MPC-44, and MPC-89 are bounded by the corresponding MPC-37, MPC-44, and MPC-89 HI-STORM FW evaluations under normal conditions of storage presented in Section 4.4 of the main chapter. Although maximum temperatures of some components, such as MPC baseplate, are higher for Version UVH analyses compared to those for standard HI-STORM FW, due to the temperature limits for individual MPC and overpack components being much higher for off-normal conditions compared to the normal storage limits, separate evaluations do not need to be performed for Version UVH system.

4.I.6.2 Accident Conditions

(a) Fire Accident

The Version UVH system contains steel inter-shell ribs that transfer heat flux due to fire more efficiently into the MPC internals, and therefore, the fire evaluations for Version UVH system are not necessarily bounded by those for standard HI-STORM FW. Therefore, a thermal evaluation of the system using CFD is performed for the most bounding canister i.e., MPC-37. An overview of the methodology is provided in the following:

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

].

The results of the evaluation are presented in Table 4.I.6.1. The results show that all the component temperatures and pressures meet their respective limits.

(b) Jacket Water Loss Accident

A description of the jacket water loss accident is presented in Section 4.6.2.2 of the main FSAR.

The maximum allowable heat load of the MPCs qualified for use in HI-STORM FW Version UVH system is significantly lower than those qualified for use in the standard HI-STORM FW version. Therefore, the component temperatures and MPC cavity pressure under jacket water loss accident condition for the MPCs qualified for use in HI-STORM Version UVH will be bounded by those presented in Section 4.6.2.2 of the main report, and thus, will be acceptable.

(c) Extreme Environment Temperatures

Following the methodology presented in Section 4.6.2.3 of the main FSAR, to evaluate the effect of extreme weather conditions, an extreme ambient temperature (Table 2.2.2 of the main FSAR) is postulated to persist for a 3-day period. [PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

].

(d) Burial Under Debris

At the storage site, no structures are permitted over the casks. Minimum regulatory distances from the storage site to the nearest site boundary precludes close proximity of vegetation. There is no credible mechanism for the Version UVH System to become completely buried under debris. However, for conservatism, a complete burial under debris scenario is considered.

Following the same methodology used for the burial-under-debris duration calculation for the standard HI-STORM FW system, the maximum allowable time for the version UVH to be buried under debris is calculated and presented in Table 4.I.6.3.

Alternatively, the CFD methodology for a burial-under-debris scenario described in Section 4.6 may be used to compute the allowable duration on a site-specific basis.

(f) Flood Accident

Many ISFSIs are located in flood plains susceptible to floods. The heat rejection of the Version UVH system to water is far more efficient than to air, and therefore, the results during normal long-term storage conditions bound those during an event of flood.

TABLE 4.I.6.1

RESULTS FOR DESIGN BASIS FIRE EVENT FOR THE MOST BOUNDING MPC-37/VERSION UVH SCENARIO

Component	Temperature, ºC (ºF)
Fuel Cladding	376 (709)
Basket	354 (669)
MPC Shell	264 (507)
MPC Lid	265 (509)
MPC Baseplate	238 (460)
Pressure, psig	
MPC Cavity	99.0
HI-STORM FW UVH Annulus	9.09

TABLE 4.I.6.2 RESULTS FOR EXTREME AMBIENT TEMPERATURE CONDITION FOR THE MOST BOUNDING MPC-37/VERSION UVH SCENARIO	
Component	Temperature, ºC (ºF)
Fuel Cladding	393 (740)
Basket	372 (701)
MPC Shell	283 (541)
MPC Lid Note 1	287 (548)
MPC Baseplate Note 1	253 (487)
Pressure, psig	
MPC Cavity	103.1
HI-STORM FW UVH Annulus	10.39
Note 1: Maximum section average temperature is reported	

TABLE 4.I.6.3
RESULTS FOR THERMAL EVALUATION OF VERSION UVH SYSTEM DURING BURIAL-UNDER-DEBRIS CONDITION

Maximum Allowable Duration (hrs)	58.9

4.I.7 Regulatory Compliance

The statements on compliance of the vented storage system to the regulatory requirements of 10CFR72 presented in Section 4.7 remain applicable to the unvented system without limitation.

4.I.8 References

[4.I.1] "Thermal Analysis of HI-STORM FW UVH System," HI-2200191, Revision 1, Holtec International (Proprietary), 2021.
[4.I.2] "Final Safety Analysis Report on the HI-STORM 100 System,' LAR 1014-16.

SUPPLEMENT 5.I

SHIELDING EVALUATION OF THE HI-STORM FW SYSTEM WITH UNVENTILATED OVERPACK

5.I.0 INTRODUCTION

This supplement presents the shielding safety evaluation of a version of the HI-STORM FW system, Version UVH, wherein the overpack's inlet and outlet air passages have been removed resulting in a complete cessation of ventilation in the space between the cask cavity and the stored multi-purpose canister (MPC) during the system's operation.

The evaluation presented herein supplements those evaluations of the HI-STORM FW 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 Version UVH 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, e.g., Section 5.I.1 correspond to Sections 5.1. Table 5.I.0.1 lists those sections that do not require any change and that are therefore omitted from this supplement.

TABLE 5.I.0.1

SECTIONS IN CHAPTER 5 OF THE FSAR THAT REMAIN APPLICABLE TO THE SAFETY EVALUATION IN SUPPLEMENT 5.I

Section number	Title	Reason for omission
5.I.2	Source specification	The information in Section 5.2 remains applicable
5.I.5	Regulatory Compliance	The information in Section 5.5 remains applicable
5.I.6	References	The information in Section 5.6 remains applicable, no new references are required

5.I.1 DISCUSSION AND RESULTS

The HI-STORM FW UVH system differs from the HI-STORM FW system evaluated in the main body of this chapter only in the use of an unventilated storage overpack. All MPCs and HI-TRAC transfer casks are identical between the systems. All calculations in this supplement are performed with limiting heat load patterns for the MPC-37, MPC-44 and MPC-89.

The principal shielding design of the HI-STORM FW UVH is identical to that of 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.

The shielding analyses were performed with MCNP5 [5.1.1], which is the same code used for the analyses presented in the main body of this chapter. The source terms methodology is developed in the main part of the report.

The zircaloy clad fuel assemblies used for calculating the dose rates presented in this supplement are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR and BWR fuel types, respectively. The same fuel assemblies are considered in the main part of the report.

Note that all results presented in Section 5.I.1 are for acceptable content (see Table 5.I.1.1) and the minimum shielding thickness of the HI-TRAC. The acceptable loading configuration is representative for the dose limits. This results in high calculated dose rates, which are presented here for illustration purposes. For the actual loading conditions both the shielding thickness and the loading content should be governed by ALARA considerations.

5.I.1.1 Normal and Off-Normal Operations

Tables 5.I.1.2, 5.I.1.3 and 5.I.1.4 provide the dose rates adjacent to and one meter from the HI-STORM FW UVH overpack during normal conditions when loaded with the MPC-37, MPC-44 and MPC-89.

Table 5.I.1.5 presents the annual dose to an individual from a single HI-STORM FW UVH cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location for the more limiting MPC. The minimum distance required for the corresponding dose is also listed. 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.

Tables 5.I.1.6, 5.I.1.7 and 5.I.1.8 provide the dose rates adjacent to and one meter from the HI-TRAC transfer cask during normal conditions when loaded with the MPC-37, MPC-44, and MPC-89.

Figure 5.I.1.1 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM FW UVH overpack.

5.I.1.2 Accident Conditions

The design basis accidents analyzed in Chapter 11 have a negligible effect on the HI-STORM FW UVH overpack, but a larger effect on the HI-TRAC, and results for this is presented in this subsection.

The dose for a period of 30 days is shown in Table 5.I.1.9 for the more limiting MPC, where 30 days is used to illustrate the radiological impact for a design basis accident.

Table 5.I.1.1

EVALUATED BURNUP, ENRICHMENTAND COOLING TIME COMBINATIONS
REPRESENTATIVE LOADING CONFIGURATION

Burnup (MWd/mtU)	**Cooling Time (years)**		
	REGION 1	REGION 2	REGION 3
MPC-37			
5000	2.0	2.0	2.0
10000	3.0	3.0	3.0
20000	5.0	4.0	4.0
30000	7.0	5.0	5.0
40000	9.0	6.0	5.0
50000	12.0	7.0	6.0
60000	22.0	9.0	8.0
70000	34.0	12.0	12.0
MPC-44			
5000	2.0	2.0	2.0
10000	5.0	3.0	3.0
20000	8.0	6.0	6.0
30000	9.0	7.0	7.0
40000	13.0	8.0	8.0
50000	19.0	11.0	9.0
60000	31.0	15.0	12.0
70000	42.0	23.0	23.0
MPC-89			
5000	1.0	2.0	2.0
10000	2.6	2.0	3.0
20000	5.0	3.0	4.0
30000	7.0	4.0	4.0
40000	8.0	5.0	5.0
50000	10.0	6.0	6.0
60000	17.0	8.0	7.0
70000	28.0	10.0	9.0

Notes:

- Regions are shown in Figures 1.2.1a, 1.2.1d, and 1.2.2.
- Minimum enrichments are shown in Table 5.0.3 for PWR and Table 5.0.4a for BWR fuel assemblies.

Table 5.I.1.2

MAXIMUM DOSE RATES FROM THE HI-STORM FW UVH OVERPACK
FOR NORMAL CONDITIONS MPC-37
REPRESENTATIVE LOADING CONFIGURATION

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 HI-STORM FW UVH						
1	40.8	0.2	0.1	<0.1	41.2	57.8
2	42.3	0.9	<0.1	0.5	43.8	61.4
3	0.3	<0.1	1.2	<0.1	1.5	3.6
4 (center)	<0.1	0.9	<0.1	1.0	2.0	2.3
4 (mid)	<0.1	1.3	0.2	0.5	2.1	3.0
4 (outer)	0.2	<0.1	0.8	<0.1	1.1	2.5
ONE METER FROM THE HI-STORM FW UVH						
1	16.2	<0.1	0.2	<0.1	16.6	23.2
2	20.0	<0.1	<0.1	<0.1	20.1	28.6
3	1.4	<0.1	0.9	<0.1	2.4	4.6
4	<0.1	0.5	0.2	0.3	0.9	1.5

Notes:

- Refer to Figure 5.I.1.1 for dose locations.
- Values in mrem/hr are rounded to the nearest tenths place.
- Dose location 3 (overpack edge) is located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is halfway between the center and outer edge of the top surface of the top lid. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.I.1.3

MAXIMUM DOSE RATES FROM THE HI-STORM FW UVH OVERPACK FOR NORMAL CONDITIONS MPC-44 REPRESENTATIVE LOADING CONFIGURATION

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 HI-STORM FW UVH						
1	48.1	<0.1	0.2	<0.1	48.3	68.5
2	51.6	<0.1	<0.1	<0.1	51.7	73.6
3	0.4	<0.1	0.8	<0.1	1.1	3.1
4 (center)	<0.1	0.8	<0.1	0.9	1.7	2.2
4 (mid)	<0.1	1.2	0.2	0.4	1.9	2.8
4 (outer)	0.3	<0.1	0.7	<0.1	1.0	2.9
ONE METER FROM THE HI-STORM FW UVH						
1	19.7	<0.1	0.3	<0.1	20.0	28.0
2	24.5	<0.1	<0.1	<0.1	24.6	34.4
3	2.0	<0.1	0.6	<0.1	2.5	5.1
4	<0.1	0.5	0.1	0.3	0.9	1.4

Notes:

- Refer to Figure 5.I.1.1 for dose locations.
- Values in units of mrem/hr are rounded to the nearest tenths place.
- Dose location 3 (overpack edge) is located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is in the middle of the top surface of the top lid. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.I.1.4

MAXIMUM DOSE RATES FROM THE HI-STORM FW UVH OVERPACK FOR NORMAL CONDITIONS MPC-89 REPRESENTATIVE LOADING CONFIGURATION

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 HI-STORM FW UVH					
1	33.9	<0.1	0.4	<0.1	34.4
2	42.6	0.1	<0.1	<0.1	42.8
3	0.2	<0.1	2.1	<0.1	2.4
4 (center)	<0.1	0.6	0.2	0.7	1.5
4 (mid)	<0.1	0.9	0.3	0.3	1.6
4 (outer)	<0.1	<0.1	0.9	<0.1	1.0
ONE METER FROM THE HI-STORM FW UVH					
1	14.0	<0.1	0.6	<0.1	14.7
2	19.4	<0.1	0.1	<0.1	19.6
3	1.3	<0.1	1.8	<0.1	3.1
4	<0.1	0.3	0.2	0.2	0.7

Notes:

- Refer to Figure 5.I.1.1 for dose locations.
- Values in units of mrem/hr are rounded to the nearest tenths place.
- Dose location 3 (overpack edge) is located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is halfway between the center and the outer edge of the top surface of the top lid. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.I.1.5

DOSE RATES FOR ARRAYS OF HI-STORM FWs Version UVH

Array Configuration	**1 cask**	**2x2**	**2x3**	**2x4**	**2x5**
REPRESENTATIVE LOADING CONFIGURATION					
Annual Dose (mrem/year)	8.8	22.4	10.1	13.5	16.8
Distance to Controlled Area Boundary (meters)	300	300	400	400	400

Notes:
- Values in units of mrem/yr are rounded to the nearest tenths place.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.

Table 5.I.1.6

MAXIMUM DOSE RATES FROM THE HI-TRAC VW WITH MPC-37
FOR NORMAL CONDITIONS
REPRESENTATIVE LOADING CONFIGURATION

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 HI-TRAC VW						
1	563.2	30.3	665.2	80.9	1339.7	1675.2
2	1902.4	108.7	<0.1	207.9	2219.0	3465.8
3	12.7	3.2	263.2	3.6	282.7	741.8
4	32.6	2.1	325.1	389.1	748.8	1474.9
5	296.2	4.2	1199.0	1702.5	3201.9	3486.1
ONE METER FROM THE HI-TRAC VW						
1	384.7	16.5	72.5	37.2	510.9	756.5
2	886.6	32.4	6.9	71.8	997.7	1593.1
3	102.3	4.4	96.8	6.6	210.1	431.3
4	38.8	0.8	199.0	109.0	347.6	767.9
5	194.2	1.3	750.8	460.4	1406.7	1569.8

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water.
- Values in units of mrem/hr are rounded to the nearest tenths place.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.I.1.7

MAXIMUM DOSE RATES FROM THE HI-TRAC VW WITH MPC-44
FOR NORMAL CONDITIONS
REPRESENTATIVE LOADING CONFIGURATION

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 HI-TRAC VW						
1	536.4	10.9	568.5	27.0	1142.9	1564.1
2	1725.0	67.2	<0.1	112.3	1904.4	3491.3
3	8.3	6.5	352.2	7.8	374.8	1212.2
4	22.3	2.8	294.0	467.4	786.4	1853.1
5	178.9	4.1	1103.5	1840.6	3127.1	3424.4
ONE METER FROM THE HI-TRAC VW						
1	367.0	1.0	71.6	2.2	441.9	801.1
2	889.8	1.6	4.1	2.7	898.3	1722.9
3	108.5	0.2	94.1	0.3	203.1	526.3
4	24.3	0.6	209.1	117.7	351.8	914.5
5	158.5	1.0	717.4	408.9	1285.9	1471.1

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water.
- Values in units of mrem/hr are rounded to the nearest tenths place.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.I.1.8

MAXIMUM DOSE RATES FROM THE HI-TRAC VW WITH MPC-89
FOR NORMAL CONDITIONS
REPRESENTATIVE LOADING CONFIGURATION

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 HI-TRAC VW					
1	226.1	26.3	2054.4	59.8	2366.6
2	2887.5	157.0	<0.1	287.7	3332.2
3	8.0	4.3	524.1	5.0	541.4
4	17.6	2.6	333.0	404.2	757.4
5	80.0	5.5	1415.2	1949.8	3450.6
ONE METER FROM THE HI-TRAC VW					
1	450.0	19.7	246.4	40.7	756.9
2	1316.7	43.1	19.5	96.2	1475.6
3	131.8	4.4	248.1	5.9	390.2
4	13.9	0.7	238.9	95.9	349.3
5	49.0	1.9	791.4	632.0	1474.3

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water.
- Values in units of mrem/hr are rounded to the nearest tenths place.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids.

Table 5.I.1.9

MAXIMUM DOSE FROM HI-TRAC VW
FOR ACCIDENT CONDITIONS
AT 100 METERS
REPRESENTATIVE LOADING CONFIGURATION

Content	Dose Rate (rem/hr)	Accident Duration (days)	Total Dose (rem)	Regulatory Limit (rem)	Time to Reach Regulatory Limit (days)
Bounding for Version UVH	3.3E-3	30	2.4	5	62

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values in units of rem are rounded to the nearest tenths place; values in units of days are rounded to the nearest integer where appropriate.
- Regulatory Limit is from 10CFR72.106.

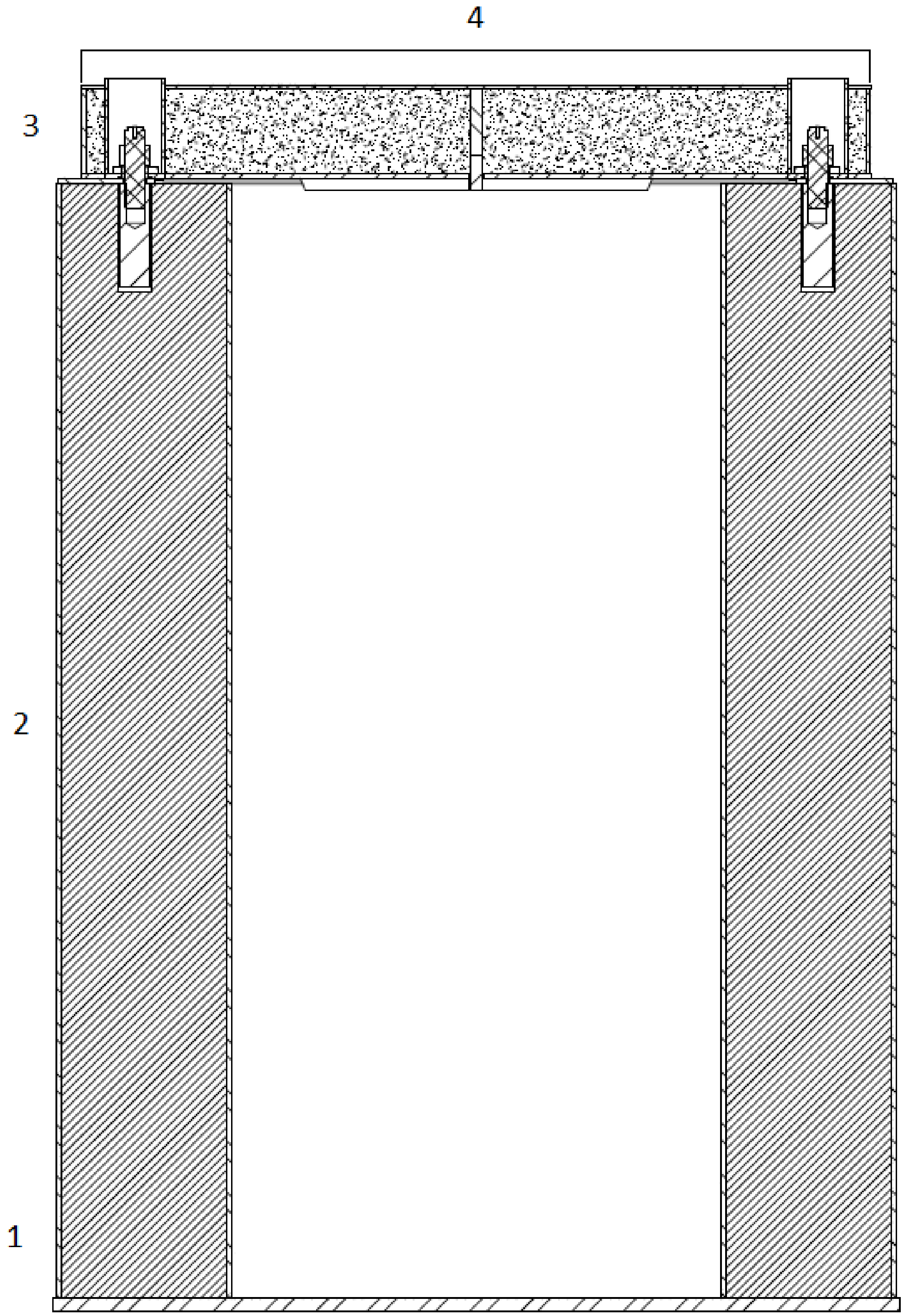

Figure 5.I.1.1

CROSS SECTION ELEVATION VIEW OF HI-STORM FW UVH OVERPACK WITH DOSE POINT LOCATIONS

5.I.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM FW UVH system was performed with MCNP5 [5.1.1], which is the same code used for the analyses presented in the main part of this chapter. A sample input file for MCNP is provided in Appendix 5.I.A.

Section 1.I.5 provides the drawings that describe the HI-STORM FW UVH 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. Figures 5.I.3.1, 5.I.3.2 and 5.I.3.3 show cross sectional views of the HI-STORM FW UVH overpack, MPCs, and basket cells as they are modeled in MCNP. Figures 5.I.3.1, 5.I.3.2 and 5.I.3.3 were created in VISED and are drawn to scale.

Composition and densities of the various materials used in the HI-STORM FW UVH system and HI-TRAC shielding analyses are given in Section 5.3.2 in the main part of the report. A minimum 3.2 g/cm^3 (200 pcf) concrete density is required for HI-STORM FW UVH for enhanced thermal conductivity as specified in Table 1.D.1 in Appendix 1.D of Reference [5.2.17]. Conservatively, 2.72 g/cm^3 (170 pcf) concrete density is used in the dose rates analysis in this Supplement. Concrete composition and density for HI-STORM FW UVH system is shown in Table 5.I.3.1.

Since the HI-STORM FW UVH 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. The MPC supports and guides were conservatively neglected.
2. The fuel shims are not modeled This is conservative since it removes steel that would provide a small amount of additional shielding.
3. The thickness of HI-STORM lid concrete is 14 inches. It is conservatively modelled as 13.75 inches.
4. Ribs of HI-STORM lid above the cover plate are not modelled. This conservatively reduces the amount of steel in the lid.

Table 5.I.3.1

CONCRETE COMPOSITION IN THE HI-STORM FW UVH SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Concrete	2.72	H	1.0
		O	53.2
		Si	33.7
		Al	3.4
		Na	2.9
		Ca	4.4
		Fe	1.4

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Figure 5.I.3.1
HI-STORM FW UVH OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

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Figure 5.I.3.2
HI-STORM FW UVH OVERPACK WITH MPC-44 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

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Figure 5.I.3.3
HI-STORM FW UVH OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

5.I.4 SHIELDING EVALUATION

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 using dose response functions. MCNP and the calculational approach used in this supplement in the same way as in the main part of this chapter in calculating the results presented in Section 5.I.1 and this section 5.I.4. An example of an MCNP input file, for the HI-STORM FW Version UVH, is listed in Appendix 5.I.A.

As discussed in Section 5.I.1, detailed results for the HI-STORM FW Version UVH overpack and HI-TRAC VW for representative loading configuration are presented in section 5.I.1. This section presents results for the HI-STORM FW UVH and HI-TRAC VW for bounding loading configuration. Bounding configuration presents an unrealistic bounding case where all cells are loaded with assemblies corresponding to the highest allowable source terms. Bounding burnup and cooling time combinations are shown in Table 5.I.4.1.

Table 5.I.4.2 provides maximum total dose rates adjacent to and one meter from the HI-STORM FW UVH overpack during normal conditions when loaded with the MPC-37, MPC-44, and MPC-89. Table 5.I.4.3 presents the annual dose to an individual from a single HI-STORM FW UVH cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location for the more limiting MPC.

Table 5.I.4.4 provides maximum dose rates adjacent to and one meter from the HI-TRAC transfer cask during normal conditions when loaded with the MPC-37, MPC-44 and MPC-89.

HI-TRAC VW is identical in design to that in the main part of this chapter, only lead thickness is increased by 1 inch to demonstrate that bounding content can be loaded.

Table 5.I.4.1
EVALUATED BURNUP, ENRICHMENTAND COOLING TIME COMBINATIONS
BOUNDING LOADING CONFIGURATION

Burnup (MWd/mtU)	**Cooling Time (years)**		
	REGION 1	REGION 2	REGION 3
MPC-37			
5000	1.0	1.0	1.0
10000	2.4	1.4	1.4
20000	3.0	2.0	2.0
30000	4.0	2.6	2.6
40000	7.0	3.5	3.5
50000	12.0	4.0	4.0
60000	22.0	5.0	5.0
70000	34.0	7.0	7.0
MPC-44			
5000	1.0	1.2	1.2
10000	3.0	1.6	1.6
20000	3.5	2.2	2.2
30000	5.0	3.0	3.0
40000	10.0	3.5	3.5
50000	19.0	5.0	5.0
60000	31.0	6.0	6.0
70000	42.0	9.0	9.0
MPC-89			
5000	1.0	1.0	1.0
10000	2.2	1.2	1.2
20000	3.0	1.6	1.6
30000	3.5	2.2	2.2
40000	5.0	2.8	2.8
50000	10.0	3.5	3.5
60000	17.0	4.0	4.0
70000	28.0	5.0	5.0

Notes:

- Regions are shown in Figures 1.2.1a, 1.2.1d, and 1.2.2.
- Minimum enrichments are shown in Table 5.0.3 for PWR and Table 5.0.4a for BWR fuel assemblies.

Table 5.I.4.2

MAXIMUM TOTAL DOSE RATES FOR THE HI-STORM FW
FOR NORMAL CONDITIONS
BOUNDING LOADING CONFIGURATION

Dose Point Location	MPC-37	MPC-44	MPC-89
ADJACENT TO THE HI-STORM FW UVH (mrem/hr)			
1	149.1	155.0	127.6
2	157.9	166.4	157.4
3	4.4	4.5	3.5
4 (center)	2.7	3.3	1.9
4 (mid)	3.5	4.2	2.1
4 (outer)	3.2	4.0	1.6
ONE METER FROM THE HI-STORM FW UVH (mrem/hr)			
1	59.9	63.5	53.3
2	73.5	77.8	72.6
3	8.7	9.5	7.7
4	1.8	2.1	1.0

Notes:

- Refer to Figure 5.I.1.1 for dose locations.
- Values in units of mrem/hr are rounded to nearest tenths place.
- Dose location 3 (overpack edge) is located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is halfway between the center and outer edge of the top surface of the top lid. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and for PWR fuel ^{60}Co from the BPRAs in the active fuel region.

Table 5.I.4.3

DOSE RATES FOR ARRAYS OF HI-STORM FWs Version UVH

Array Configuration	**1 cask**	**2x2**	**2x3**	**2x4**	**2x5**
BOUNDING LOADING CONFIGURATION					
Annual Dose (mrem/year)	19.3	17.9	10.1	13.5	16.8
Distance to Controlled Area Boundary (meters)	300	400	500	500	500

Notes:
- Values in units of mrem/yr are rounded to nearest tenths place.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.

Table 5.I.4.4

MAXIMUM TOTAL DOSE RATES FOR THE HI-TRAC VW
FOR NORMAL CONDITIONS
BOUNDING LOADING CONFIGURATION

Dose Point Location	**MPC-37**	**MPC-44**	**MPC-89**
ADJACENT TO THE HI-TRAC VW (mrem/hr)			
1	819.1	940.4	1075.6
2	2101.0	2256.4	2988.5
3	198.8	362.2	186.6
4	1818.0	2564.6	1138.7
5	5119.5	6287.5	5207.8
ONE METER FROM THE HI-TRAC VW (mrem/hr)			
1	443.7	470.6	551.5
2	957.4	1020.7	1360.9
3	176.7	216.5	195.8
4	917.6	1228.2	559.9
5	2266.7	2743.4	2334.7

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water.
- Values in units of mrem/hr are rounded to nearest tenths place.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and for PWR fuel ^{60}Co from the BPRAs in the active fuel region.

APPENDIX 5.I.A

SAMPLE MCNP INPUT FILE FOR HI-STORM FW UVH

Proprietary Appendix Withheld in Accordance with 10 CFR 2.390

CHAPTER 9.I: OPERATING PROCEDURES

9.I.0 INTRODUCTION

The operations associated with the use of the HI-STORM FW UVH system, described in Supplement 9.I, are like the operations for the standard HI-STORM FW system. The following sections describe those operations that are, in any respect, unique to the HI-STORM FW UVH system and thus supplement the information presented in Chapter 9. Where practical, the section numbers used below directly references the corresponding section in Chapter 9. For example, Subsection 9.I.2.6 supplements the operations described in Subsection 9.2.6. The guidance provided in this supplement shall be used along with the operations procedures provided in Chapter 9 to develop the site-specific operating procedures for the HI-STORM FW UVH.

9.I.1 TECHNICAL AND SAFETY BASIS FOR LOADING AND UNLOADING PROCEDURES

The Technical and Safety Basis for loading and unloading the HI-STORM FW identified in Section 9.1 of Chapter 9 are applicable to the HI-STORM FW UVH.

9.I.2 PROCEDURE FOR LOADING THE HI-STORM FW UVH SYSTEM IN THE SPENT FUEL POOL

The procedures presented within Subsections 9.2.1 through 9.2.5 of Chapter 9 are identical for the HI-STORM FW UVH system. The changes to operations when placing the HI-STORM FW UVH into storage are described below.

9.I.2.6 Placement of HI-STORM FW UVH into Storage

The following instructions shall be incorporated to the cask operations as additional steps to the generic guidance in Section 9.2.6 on loading operations for unventilated cask models in Chapter 9:

1. Before installing the Closure Lid on the cask body, the lid gasket is placed on the top of the cask's top ring.

2. Inspect cask cavity and confirm to be visibly dry (free of standing water).

3. Place cask lid on top of the gasket.

4. Continue with the steps of Subsection 9.2.6 of Chapter 9 for conducting the required surface dose rate measurements in accordance with the Technical Specification and movement of the overpack to its storage location on the ISFSI pad.

5. After the cask is placed in its storage location on the ISFSI pad, install lid studs, washers, and hex nuts onto the cask.

6. Tighten lid hex nuts to the point of contact with the washer. Then loosen nut to provide a nominal axial gap of 0.5".

7. If the site is using non-oxidizing gas in place of air in the annulus, evacuate air in the MPC/HI-STORM FW UVH annulus and replace with dry nitrogen (or another non-oxidizing gas) using couplings provided in the small penetrations in the cask body. The target fill pressure of the non-oxidizing fill gas shall be determined on a site-specific basis to meet the design pressure indicated in Table 4.I.1.3.

Table 9.I.2.1

HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
HI-STORM UVH Annulus Evacuation System	Not Important To Safety	Used to evacuate air from the HI-STORM UVH annulus space.
Nitrogen (or another non-oxidizing gas) Backfill System	Not Important To Safety	Used for controlled insertion of nitrogen into the HI-STORM UVH for placement into storage.

Table 9.I.2.2

HI-STORM FW SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Instrument	Function
Pressure Gauges	Ensures correct pressure during HI-STORM backfill operations.

† All instruments require calibration.

Table 9.I.2.3 HI-STORM FW UV SYSTEM OVERPACK INSPECTION CHECKLIST
Note: This checklist provides a supplement to the main table 9.2.3 as a basis for establishing additional steps to a site-specific inspection checklist for the HI-STORM FW UVH overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation, and potential corrective action prior to use.

<u>HI-STORM FW UVH Overpack Lid:</u>

1. Lid sealing surfaces shall be cleaned and inspected for corrosion, scratches, and gouges.
2. Lid seal shall be inspected for cuts, abrasions, or other damage which may affect is function.
3. Vent and vent screen inspections are not required because the HI-STORM FW UVH lid does not include vents.

<u>HI-STORM FW UVH Main Body:</u>

1. Vent and vent screen inspections are not required because the HI-STORM FW UVH body does not include vents.

9.I.3 ISFSI OPERATIONS

The HI-STORM FW UVH system heat removal system is a totally passive system. Maintenance on the HI-STORM FW UVH system is typically limited to cleaning and touch-up painting of the overpacks. The HI-STORM FW UVH system does not have vents which require surveillance. In the unlikely event of significant damage to the HI-STORM FW UVH, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM FW UVH overpack. The procedures in Section 9.2 should be used to reposition a HI-STORM FW UVH overpack for minor repairs and maintenance. In extreme cases, Section 9.I.4 provides guidance in addition to Section 9.4 of Chapter 9 for unloading the MPC from the HI-STORM FW UVH.

9.I.4 PROCEDURE FOR UNLOADING THE HI-STORM FW UVH FUEL IN THE SPENT FUEL POOL

The HI-STORM FW UVH system unloading procedures shall be identical to main chapter section 9.4.1. 9.4.3,9.4.4 and 9.4.5. Additional steps for section 9.4.2 are included below for the removal of the HI-STORM UVH lid.

9.I.4.2 HI-STORM FW UVH Recovery from Storage

1. Prior to recovering the MPC from HI-STORM FW UVH, the following step shall be performed:

   Vent HI-STORM FW UVH to atmospheric pressure, such that the interior pressure is equal to ambient pressure prior to any HI-STORM FW UVH lifting operations.

CHAPTER 10.I†: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

10.I.0 INTRODUCTION

The acceptance and maintenance program associated with the use of the HI-STORM FW UVH system, described in Supplement 10.I, are like the maintenance and acceptance program for the standard HI-STORM FW system. The following sections describe the requirements that are, in any respect, unique to the HI-STORM FW UVH system and thus supplement the information presented in Chapter 10. Where practical, the section numbers used below directly reference the corresponding sections in Chapter 10. For example, Subsection 10.I.2 supplements the requirements described in Subsection 10.2. The guidance provided in this supplement shall be used along with the procedures provided in Chapter 10 to develop the site-specific maintenance program for the HI-STORM FW UVH.

10.I.1 ACCEPTANCE CRITERIA

10.I.1.1 Fabrication and Nondestructive Examination (NDE)

The HI-STORM FW UVH does not introduce any new fabrication or NDE requirements.

10.I.1.2 Structural and Pressure Tests

The HI-STORM FW UVH does not introduce any new structural or pressure test beyond what is presented in Subsection 10.1.2. Pressure testing of the HI-STORM FW UVH Body is not required due to low operating pressure.

10.I.1.3 Materials Testing

There are no new structural and shielding materials used for the HI-STORM FW UVH. No additional materials testing is required for the HI-STORM FW UVH. The HI-STORM Lid seal will be manufactured from a metallic material that is demonstrated to have good radiation resistance such that degradation over the service life of the cask is not a concern.

10.I.1.4 Leakage Testing

There is no leakage test required for the HI-STORM FW UVH boundary. The function of the HI-STORM FW UVH seal is to provide a barrier against deleterious effects of the environment, not as a pressure boundary. The only requirement is that the gasket is inspected to ensure that it is intact and new before the lid is installed

10.I.1.5 Component Tests

10.I.1.5.1 Valves, Pressure Relief Devices, and Fluid Transport Devices

There are no additional valves or pressure relief devices introduced for the HI-STORM FW UVH System. Excess pressure is released from the boundary by the HI-STORM lid momentarily lifting from the body and then re-seating.

10.I.1.5.2 Seals and Gaskets

The Lid to Cask body in the unventilated overpack features a gasket to isolate the environment in the cask's cavity space from ambient air. The gasket does not perform a safety significant function and thus no additional testing is required

10.I.1.6 Shielding Integrity

There are no new tests or inspections required for shielding integrity.

10.I.1.7 Thermal Acceptance Tests

There are no new tests or inspections required for thermal acceptance.

10.I.1.8 Cask Identification

There are no new marking requirements.

10.I.2 MAINTENANCE PROGRAM

As the addition of the unventilated overpack through Supplement # I does not involve the introduction of any new structural or shielding materials, MPCs or transfer cask to the Storage system, only minimal changes to the Maintenance activities outlined in Section 10.2 of Chapter 10 are required. Any additional tests, inspections, and maintenance activities are identified in the following Subsections.

10.I.2.1 Structural and Pressure Parts

No additional maintenance for structural and pressure parts is required for the HI-STORM FW UVH.

10.I.2.2 Leakage Tests

Leakage tests are not a requirement for the storage maintenance program.

The unventilated Storage system lid gasket requires the additional maintenance step of replacement anytime the joint is completely disassembled. A new gasket shall be used upon re-assembly.

10.I.2.3 Subsystem Maintenance

The HI-STORM FW UVH does not have vents and will not have the option a monitoring system which must be maintained.

10.I.2.4 Pressure Relief Devices

There is no additional pressure relief device introduced for the HI-STORM FW UVH System which must be maintained.

10.I.2.5 Shielding

There are no additional shielding maintenance requirements for the HI-STORM FW UVH.

10.2.6 Thermal

The HI-STORM FW UVH does not include air vents. As a result, surveillance or monitoring is not required during storage operations.

Table 10.I.2.1

HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	**Frequency**
HI-STORM UVH Lid Seal Replacement	In the event the HI-STORM lid is completely disassembled from the body.

10.I.3 REGULATORY COMPLIANCE

There are no additional requirements for the HI-STORM FW UVH.

SUPPLEMENT 11.I: RADIATION PROTECTION

11.I.0 INTRODUCTION

This supplement discusses the design considerations and operational features that are incorporated in the HI-STORM FW system, Version UVH to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, transfer, and on-site dry storage.

The evaluation presented herein supplements those evaluations of the HI-STORM FW overpacks contained in the main body of Chapter 11 of this FSAR, and information in the main body of Chapter 11 that remains applicable to the Version UVH 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, e.g., Section 11.I.0 correspond to Sections 11.0. Table 11.I.0.1 lists those sections that do not require any change and that are therefore omitted from this supplement.

TABLE 11.I.0.1

SECTIONS IN CHAPTER 11 OF THE FSAR THAT REMAIN APPLICABLE TO THE SAFETY EVALUATION IN SUPPLEMENT 11.I

Section number	Title	Reason for omission
11.I.1	ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)	The information in Section 11.1 remains applicable
11.I.5	REFERENCES	The information in Section 11.5 remains applicable, no new references are required

11.I.2 RADIATION PROTECTION FEATURES IN THE SYSTEM DESIGN

The design of the HI-STORM FW UVH components has been principally focused on maximizing ALARA during the short-term operations as well as during long-term storage. Some of the key design features engineered in the system components to minimize occupational dose and site boundary dose are summarized in Table 11.I.2.1.

The key design features engineered in the HI-TRAC VW components to minimize occupational dose and site boundary dose are summarized in Table 11.2.1 of Chapter 11.

Table 11.I.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE

#	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
1.	HI-STORM FW UVH Overpack	Use of the steel weldment structure permits the density of concrete (set at a minimum of 170 lb/cubic feet) to be increased to maximum values specified in Table 1.2.5.	A
2.	HI-STORM FW UVH Overpack	Cask's vertical disposition and use of a thick lid (see drawing package in Section 1.I.5) and high density concrete minimizes skyshine.	B
3.	HI-STORM FW UVH Overpack	HI-STORM FW UVH overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance degradation (e.g., corrosion).	A
4.	HI-STORM FW UVH Overpack	HI-STORM FW UVH has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel (see Section 1.4).	A
5.	HI-STORM FW UVH Overpack	The steel structure of the HI-STORM FW UVH overpack gives it the fracture resistance properties that protect the overpack from developing streaming paths in the wake of the impact from a projectile such as a tornado missile strike or handling incident.	A, B
6.	HI-STORM FW UVH Overpack	Absence of vent openings eliminates localized areas of increased dose rates from the source of radiation emitted from the Canister to workers during loading and handling operations. The absence of vent openings also reduces dose rates from stationary casks on an ISFSI and minimizes dose rates to nearby buildings, to the nearby environment, and to the site boundary. The need for periodic inspection of the vent openings and associated LCOs in the CoC becomes unrequired, eliminating this source of radiation dose to the site staff.	A, B

11.I.3 ESTIMATED ON-SITE CUMULATIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM FW UVH system. This section uses the shielding analysis provided in Supplement 5.I, and the operations procedures provided in Chapter 9.

To provide a uniform basis for the dose estimates presented in this chapter, the MPC contents data considered, available HI-TRAC VW weight, etc., are provided in Table 11.I.3.1.

Apart from the operational considerations, the assumptions with regards to the cask content have a significant impact on the estimated dose. The cask loading content is very conservative, i.e., in an attempt to indicate what the highest expected dose rates could possibly be. The loading content is summarized in Table 5.I.1.1.

Using Table 11.I.3.1 data, the dose data for fuel loading (wet to dry storage) is provided in Table 11.I.3.2.

11.I.3.1 Estimated Exposures for Loading and Unloading Operations

Exposures estimates presented in Tables 11.I.3.2 is expected to bound those for unloading operations. This assessment is based on the similarity of many loading steps versus unloading operations with the elimination of several of the more dose intensive operations (such as weld inspections and leakage testing). Therefore, loading estimates should be viewed as bounding values for the contents considered for unloading operations.

11.I.3.2 Estimated Exposures for Surveillance and Maintenance

Table 11.3.3 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI.

Table 11.I.3.1 ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE		
#	Item	Value
1.	MPC-Contents (MPC-37)[1]	See Table 5.I.1.1
2.	Weight of HI-TRAC VW Full of Fuel and Water	125 tons
3.	HI-STORM FW UVH Concrete Density	170 lb/cubic feet

[1] The case of MPC-37 is used but similar results are expected for all MPC types.

TABLE 11.I.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW UVH SYSTEM				
Task Description	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
Fuel loading and removal of the transfer cask and MPC from the spent fuel pool (includes: fuel loading, fuel assembly identification check, MPC lid installation, Lift Yoke attachment to the HI-TRAC VW, HI-TRAC VW removal from the spent fuel pool, preliminary decontamination, HI-TRAC VW movement to the DAS, Lift Yoke removal and decontamination). Background radiation of 1 mrem/hr assumed.	3	800	1.0	40.0
MPC preparation for closure (includes: HI-TRAC VW and MPC decontamination, radiation surveys, partial MPC pump down, annulus seal removal, partial lowering of annulus water level, annulus shield ring installation, weld system installation); workers assumed to be on scaffolding near the top of the HI-TRAC.	3	30	181.9	272.9
MPC Closure (includes MPC lid to shell welding, weld inspection). Assumes welding machine uses standard Holtec pedestal which provides additional shielding. Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	2	185	181.9	112.2

TABLE 11.I.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW UVH SYSTEM				
Task Description	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
MPC Preparation for Storage (includes: MPC hydrostatic testing, draining, drying and backfill, vent and drain port cover plate installation, welding, weld inspection and leakage testing). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 20% of the total duration.	2	170	431.3	488.8
MPC Closure Ring Installation (includes: closure ring to MPC shell welding, weld inspection and leakage testing of the MPC primary closure). Holtec auxiliary shielding methods and equipment assumed (lead blankets, water shields, etc.) Assumes operators are present for 10% of the total duration.	2	80	854.0	227.7
HI-STORM FW UVH system preparation for receiving MPC (includes: HI-STORM FW UVH overpack positioning at transfer location, HI-STORM lid removal, Mating Device installation on HI-STORM FW UVH overpack).	3	160	0 / 0	0 / 0

TABLE 11.I.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW UVH SYSTEM				
Task Description	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
MPC Transfer (attachment of MPC lifting device, movement of HI-TRAC VW to transfer location, placement of HI-TRAC VW in Mating Device, bottom lid removal, MPC lowering, HI-TRAC VW removal, MPC lift device removal). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	3	120	463.5	278.1
HI-STORM FW UVH overpack movement to the ISFSI (will include: movement of the HI-STORM FW overpack from the fuel building to placement of the HI-STORM FW overpack on the ISFSI pad, disconnecting transporter, attachment of HI-STORM FW lid, attachment of thermal monitoring system). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 50% of the total duration.	3	220	22.5	123.8
TOTAL EXPOSURE (person-mrem)	1543.4			

11.I.4 ESTIMATED CONTROLLED AREA BOUNDARY DOSE ASSESSMENT

11.I.4.1 Controlled Area Boundary Dose for Normal Operations

Table 5.I.4.3 presents dose rates at various distances from sample ISFSI arrays for the bounding burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Supplement 5.I. 10CFR72.106 [11.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore, this is the minimum distance analyzed in Supplement 5.I. One hundred percent (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 are presented only as an illustration to demonstrate that the HI-STORM FW UVH system is in compliance with 10CFR72.104 [11.0.1]. 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 [11.0.1] contributors and 10CFR20 [11.1.1].

11.I.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Supplement 12.I, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, and off-normal handling of HI-TRAC VW) do not result in the degradation of the HI-STORM FW UVH 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.

11.I.4.3 Controlled Area Boundary Dose for Accident Conditions

The worst case shielding consequence of the accidents evaluated in Supplement 12.I for the loaded HI-TRAC VW transfer cask assumes that as a result of a fire, tornado missile, or handling accident, that all the water in the water jacket is lost or that all Holtite-A in the Neutron Shield Cylinder is lost. The shielding analysis of the HI-TRAC VW with complete loss of the water from the water jacket is discussed in Subsection 5.1.1. The results in that subsection show the resultant dose rate at the 100-meter controlled area boundary during the accident condition. At the calculated dose rate, Table 5.I.1.9 shows the calculated time to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Supplement 12.I. Therefore, the dose requirement of 10CFR72.106 [11.0.1] is satisfied. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

CHAPTER 12.I: OFF-NORMAL AND ACCIDENT EVENTS

12.I.0 Introduction

In this chapter, the off-normal and accident events germane to the HI-STORM FW UV system are considered. Because no new MPC or transfer cask are introduced in Chapter I, the off-normal and accident events applicable to them remain unchanged and therefore, are not required to be evaluated herein. Furthermore, events resulting from vent openings in the overpack are also not applicable for the ventless UV overpack. Finally, a survey of the regulatory literature shows that the unvented overpack does not introduce any new off-normal or accident event of safety consequence[1]. Therefore, the number of events that merit consideration in this chapter is vastly reduced. Those events that are applicable to the unvented overpack are evaluated in the following.

[1] The case of leakage of the gasket in the overpack is included even though it is not a safety significant event.

12.I.1 Off-Normal Conditions

The applicable off-normal events are:

i. Elevated Off-normal environmental temperature – The off-normal ambient condition case of -40°F is important only for consideration of protection against brittle fracture for which the Storage System has been qualified in Chapter 3 and so stated in Chapter 12. This conclusion remains valid because the type of materials used and their thicknesses have not been changed in Supplement I.

ii. Leakage of the Lid to overpack seal.

12.I.1.1[1] Off-Normal Environmental Temperature

The elevated off-normal temperature condition is evaluated against the off-normal condition temperature limit for the Storage system components listed in Table 2.2.3 of the main chapter.

12.I.1.1.1 Postulated Cause of Off-Normal Environmental Temperature

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. As in the main chapter, to determine the effects of the off-normal environmental temperature, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the Storage System to achieve thermal equilibrium. Because of the large mass of the Storage System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

12.I.1.1.2 Detection of Off-Normal Environmental Temperature

The analysis in Chapter 4.I shows that the Storage System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. Therefore, there is no safety imperative for detection of off-normal environmental temperatures.

12.I.1.1.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperature

- Structural: The rise in the ambient temperature will cause an increase in the cask cavity pressure which, as calculations in Chapter 4.I show, will reduce the extent of sub-atmospheric condition inside the cask which directly reduces the stress in the cask structure. However, conservatively bounding pressures under normal conditions are used in structural evaluation of cask in Chapter 3.I which envelope the off-normal ambient temperature condition with regards to the state of stress in the cask structure.

- Thermal: Thermal analysis summarized in Chapter 4.I shows that temperature of all components remains below their respective limits.

[1] The numbering of the events follows that in the main chapter with the Roman numeral I inserted to indicate that it is a part of the chapter.

- 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 rendered by the Storage System's MPC as a result of this off-normal event.
- Radiation Protection: Since there is no degradation in shielding or confinement capabilities of the Storage System, there is no effect on occupational or public exposures as a result of this off-normal event.

12.I.1.1.4 Corrective Action

Because elevated ambient temperature is a natural event and does not impair the compliance of the Storage system with the acceptance criteria set forth in Chapter 2 and Chapter 2.I, no remedial action is required.

12.I.1.1.5 Radiological Impact:

There is no radiological impact from the elevated ambient temperature on the Storage System.

Based on the above evaluation, it is concluded that the elevated off-normal temperature event does not affect the safe operation of the Storage System.

12.I.1.2 Leakage of One Seal

12.I.1.2.1 Postulated cause

Long term exposure to varying weather conditions can degrade the polymeric gasket resulting in air in-leakage in the cask and causing its sub-atmospheric cavity pressure to begin approaching the ambient.

12.I.1.2.2 Detection of leakage

Air in-leakage does not impact the safety function of the cask; therefore, there is no safety-driven imperative to detect leakage. However, monitoring of the pressure in the cask's cavity provides the means to infer air leakage into the cask

12.I.1.2.3 Effects and consequences of seal failure

MPCs are designed to be exposed to ambient air. Therefore, there is no adverse impact on the Storage System if the pressure inside the cask cavity were to rise all the way up to the ambient, as explained below:

- Structural: The rise in the cask cavity pressure to the ambient reduces the extent of sub-atmospheric condition inside the cask which directly reduces the stress in the cask structure. Hence, intrusion of air into the cask cavity would ameliorate the state of stress in the cask structure. In addition, conservatively bounding pressures under normal conditions are used in structural evaluation of cask in Chapter 3.I which envelope the off-normal condition with regards to the state of stress in the cask structure.

- Thermal: Since the safety analysis in Chapter 4.I uses the conductivity of air, there is no reduction in the computed thermal margin due to in-leakage of air.

- 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 rendered by the Storage System's MPC as a result of this off-normal event.

- Radiation Protection: Since there is no degradation in shielding or confinement capabilities of the Storage System, there is no effect on occupational or public exposures as a result of this off-normal event.

12.I.1.2.4 Corrective Action:

While the loss of seal does not affect the System's safety function, replacement of gasket shall be carried out upon discovery to restore the non-oxidizing gas environment inside the cask.

12.I.1.2.5 Radiological Impact:

There is no radiological impact from the in-leakage of air in the cask's cavity.

Based on the above evaluation, it is concluded that the off-normal event resulting in the loss of seal effectiveness does not affect the safe operation of the Storage System.

12.I.2 Accident Events

The accident events germane to the introduction of the unvented overpack in the Storage System excerpted from Table 12.2.1 are summarized in Table 12.I.2 where those requiring a detailed evaluation are shown in italicized text.

12.I.2.1 Design Basis Fire

The fire accident under on-the-pad storage is conservatively postulated in Subsection 4.6.2 of the main report. The acceptance criteria for the fire accident are provided in Subsection 2.2.3.

12.I.2.1.1 Postulated cause:

Fire in the cask transporter visiting the ISFSI pad is a probable cause for fire.

12.I.2.1.2 Detection:

The fire at the ISFSI equipped with smoke detectors is easy to detect.

12.I.2.1.3 Analysis of effects and consequences:

The thermal model described in Section 4.I is utilized to quantify the effect of the Design Basis Fire. The transport vehicle fuel tank fire has been analyzed to evaluate the outer layers of the storage overpack heated by the incident thermal radiation and forced convection heat fluxes and to evaluate fuel cladding and MPC temperatures.

- *Structural*: There are no structural consequences as a result of the fire accident condition since the accident temperature limit of the concrete is not exceeded and all component temperatures remain within applicable temperature limits (Table 2.2.3). The accident condition pressure evaluations for cask in Chapter 3.I bound the fire accident condition. The MPC structural boundary remains within accident condition internal pressure and temperature limits.

- *Thermal*: Based on a conservative analysis discussed in Chapter 4.I, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the Storage System to maintain cooling of the spent nuclear fuel within the ISG-11 Rev 3 temperature limits (Table 2.2.3) during and after fire is not compromised.

- *Shielding*: Because the shielding concrete remains below its accident temperature limit, there is no adverse effect on the shielding function 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 since the structural integrity of the confinement boundary is unaffected.

- *Radiation Protection*: Since there is minimal reduction, if any, in the cask's shielding capacity 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 the above evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the Storage System.

12.I.2.2 Non-mechanistic Tip-over

The freestanding Version UVH storage overpack, containing a loaded MPC, cannot tip-over as a result of postulated natural phenomenon events, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a *non-mechanistic* tip-over scenario per NUREG-1536 is analyzed per Paragraph 2.2.3b in Subsection 3.I.3.8 following the same methodology that was employed in the system's original certification documented in Subparagraph 3.4.4.1.4.

12.I.2.2.1 Cause of Tip-Over

The tip-over accident is stipulated as a non-mechanistic accident because a credible mechanism for the cask to tip over cannot be identified. Detailed discussions are provided in Subsections 3.1.2 and 3.4.4.

However, it is recognized that the mechanical loadings at a specific ISFSI may be sufficiently strong to cause a tip-over event, even though such a scenario is determined to be counterfactual under the loads treated in this FSAR. To enable the safety evaluation of a postulated tip-over scenario, it is necessary to set down an analysis methodology and the associated acceptance criteria. In Paragraph 2.2.3b, Subparagraph 3.4.4.1.4 and Subsection 3.I.3.8, the methodology and acceptance criteria are presented and reference tip-over problems are solved with MPC-37, MPC-44 and MPC-89 baskets. The reference tip-over problems correspond to a free rotation of the Version UVH overpack from the condition of rest at the incipient tipping point (i.e., C.G.-over-corner). The evaluations presented below refer to the above non-mechanistic tip-over scenario.

12.I.2.2.2 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 described in Subsection 3.I.3.8. The structural analysis demonstrates the following:

(i) The lateral plastic deformation of the basket panels in the active fuel region is less than the limiting value in Table 2.2.11.
(ii) The plastic strains in the MPC enclosure vessel remain below the allowable material plastic strain limit.
(iii) The cask closure lid does not dislodge after the tipover event, i.e., the closure lid bolts remain in-tact.
(iv) The lid or the cask body do not suffer any gross loss of shielding.

The side impact will cause some localized damage to the concrete and outer shell of the overpack in the local area of impact. However, there is no significant adverse effect on the structural, confinement, thermal, or criticality performance.

As mentioned earlier the non-mechanistic tip-over accident is addressed to demonstrate the defense-in-depth features of the design.

12.I.2.2.3 Tip-over Accident Corrective Actions

Corrective action after a tip-over would include a radiological and visual inspection to determine the extent of the damage to the overpack and the contained MPC. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

TABLE 12.I.1

ACCIDENT CONDITION EVENTS

Event	**Location in the main report**	**Comment (Cases that are italicized have been determined to require complete evaluation which is provided in the subsections above)**
Overpack handling accident	12.2.2	The unvented overpack has the same handling characteristics as the vented type. Therefore, the discussion in subsection 12.2.2 applies.
Non-mechanistic tip-over	*12.2.3*	*The unvented overpack requires addition evaluation because the unvented overpack has a different lid design, the MPC guide tubes are eliminated and the clearance between MPC and overpack is reduced. This is evaluated in Subsection 12.I.2.2.*
Design Basis Fire	*12.2.4*	*This condition requires additional evaluation because the unvented overpack is thermally more conductive and hence more responsive to fire. This is evaluated in Subsection 12.I.2.1.*
Tornado borne missiles	12.2.6	The unvented overpack has improved tornado missile resistance in the absence of vent openings. The evaluation in Subsection 3.I.3.7 also demonstrates that the maximum cask displacements (sliding and rocking) are governed by the results for standard FW cask. Therefore, the safety justification in Subsection 12.2.6 applies.
Design Basis Flood	12.2.7	A vulnerability in the vented models, the unvented overpack does not suffer from a deleterious scenario such as "smart flood". Furthermore, the heat rejection rate to the flood waters will be greater. In addition, the evaluation in Subsection 3.I.3.4 demonstrates that the maximum flood water velocity established in Subparagraph 3.4.4.1.1 remains governing for Version UVH cask. Therefore, a flood event does not challenge the safety performance of the Storage System containing an unvented overpack.
Earthquake	12.2.8	The discussion and approach to deal with earthquake in Chapters 2, 3 and 12 applies to the unvented overpack-bearing storage system without any modification. Additional information is provided in Subsection 3.I.3.6.
Explosion	12.2.11	The discussion and approach to deal with an explosion event, discussed in Subsection 12.2.11, applies to the unvented overpack-bearing storage system. In addition, the discussion in Section 2.I.2 regarding AEP is applicable. Additional information is provided in Subsection 3.I.3.5.
Lightning	12.2.12	As discussed in Subsection 12.2.12, lightning is an inconsequential event to the Storage System.
Burial-under-debris	12.2.14	Since the standard HI-STORM FW is primarily cooled by ventilation while the Version UV system is not, a burial-under-debris accident will have a much more significant impact on the temperatures for the standard version. A standard HI-STORM FW without ventilation is thermally equivalent to the Version UV system. Since the maximum allowable heat load for Version UV system is significantly lower than that for the standard version, therefore, the evaluation for the standard version bounds that for Version UV

Extreme Environmental Temperature	12.2.15	The consideration of elevated off-normal temperature in Subsection 12.I.1.1 in the foregoing applies without any change to the accident condition case.