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).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs), neutron source assemblies (NSAs), or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM FW system has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

As described in Chapter 1, the packaging of fuel in all HI-STORM FW MPCs will follow the heat load limitation of approximately 47 kW for the MPC-37 (see Table 1.2.3) and 46 kW for the MPC-89 (see Table 1.2.4).

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 FW System offers a three-region regionalized loading configuration in the MPC-37 and MPC-89, as described in Section 2.1. This regionalized storage pattern is guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles.

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

† 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).

Acceptance Criteria

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

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

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

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR72.106.

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

Consistent with the Standard Review Plan, NUREG-1536, this chapter contains the following information:

- A description of the shielding features of the HI-STORM FW system, including the HI-TRAC transfer cask.
- A description of the source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM FW system, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- Analyses to show that the 10CFR72.106 controlled area boundary radiation dose limits can be met during accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters. Since only representative dose rate values for normal conditions are presented for this chapter, compliance with the

radiation and exposure objectives of 10CFR72.104 is not being evaluated herein but will be performed as part of the site specific evaluations.

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

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

Chapter 11, Radiation Protection, contains the following information:

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

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM FW system are:

- Gamma radiation originating from the following sources:

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

- Neutron radiation originating from the following sources

  1. Spontaneous fission
  2. α,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4. γ,n reactions (this source is negligible)

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the stainless steel structure and the basket of the MPC and the steel, lead, and water in the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete ("Metcon" structure) of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations.

The shielding analyses were performed with MCNP5 [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR and BWR fuel types, respectively. Required site specific shielding evaluations will verify whether those assemblies and assembly parameters are appropriate for the site-specific analyses. Subsection 2.1 specifies the acceptable fuel characteristics, including the acceptable maximum burnup levels and minimum cooling times for storage of fuel in the HI-STORM FW MPCs.

The following presents a discussion that explains the rationale behind the burnup and cooling time combinations that are evaluated in this chapter for normal and accident conditions.

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 dose rates, next to the casks and at various distances, to aid the user in the applying ALARA considerations and planning of the ISFSI. To that extent, it is sufficient to present reasonably conservative dose rate values, based on a reasonable conservative choice of burnups and cooling times of the assemblies.
- 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.

As stated in Section 5.0, the HI-STORM 100 FW System offers a three-region regionalized loading configuration. Based on this configuration, fuel assemblies with higher heat loads would be loaded in the inner region allowing the user to take advantage of self-shielding from fuel assemblies with lower heat loads in the outer regions (see Tables 1.2.3 and 1.2.4). A more detailed description of the benefits of regionalized loading can be found in Section 5.4. However, for simplification, the shielding analyses are performed for a single region, i.e. assuming all assemblies in the basket have the same burnup and cooling time. The burnup and cooling time combination is selected as a representative average for the entire basket. This way, dose rates on the outer radial surface of the cask and at distances from the cask will be slightly overestimated, since the basket periphery has assemblies with below average burnup and/or above average cooling times. For the top of the cask, the average burnup and cooling time is expected to result in more realistic dose rate. The representative burnup and cooling time is then selected based on the other limiting fuel parameter, namely the average heat load per assembly. 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 increased dose values, due to the non-linear relationship between burnup and neutron source term. On the other hand, for very short cooling times, with corresponding lower burnups, dose locations that are gamma-dominated may show increased 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.

Based on these considerations, average burnup and cooling time values are selected for all calculations for normal conditions, i.e values that are away from the extreme values. The selected values are shown in Table 5.0.1, and are based on a total heat load of approximately 46 kW for the MPC-37 and 42 kW for the MPC-89. For the accident conditions however, it is recognized that the bounding accident condition is the loss of water in the HI-TRAC VW, a condition that is neutron dominated due to the removal of the principal neutron absorber in the HI-TRAC VW (water). For this case, the upper bound burnup is selected, in order to maximize the neutron source strength of all assemblies in the basket, and a corresponding higher cooling time is selected in order to meet the overall heat load limit in the cask. The resulting burnup and cooling times values for accidents are therefore different from those for normal conditions and are listed in Table 5.0.2. In all cases, low initial enrichments are selected, which further increases the neutron source terms from the assemblies

With the burnup and cooling times selected based on above considerations, dose rates calculated for normal conditions will be reasonably conservative, while for accident conditions those will represent reasonable upper bound limits.

Table 5.0.1

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR NORMAL CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
MPC-37	**MPC-89**
45,000 MWD/MTU 5 Year Cooling 3.6 wt% U-235 Enrichment	45,000 MWD/MTU 5 Year Cooling 3.2 wt% U-235 Enrichment

Table 5.0.2

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR ACCIDENT CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
MPC-37	**MPC-89**
65,000 MWD/MTU 10 Year Cooling 4.8 wt% U-235 Enrichment	65,000 MWD/MTU 10 Year Cooling 4.8 wt% U-235 Enrichment

5.1.1 Normal and Off-Normal Operations

Chapter 12 discusses the potential off-normal conditions and their effect on the HI-STORM FW system. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

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

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

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

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 11 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system in Section 2.3.5.2 as: 150 mrem/hour on the radial surface of the overpack, 250 mrem/hour at the openings of the air vents, and 60 mrem/hour on the top of the overpack.

Figure 5.1.1 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM FW overpack. Dose Point #2 is located on the side of the cask at the axial mid-height. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the dose location on the overpack lid. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width.

Figure 5.1.2 identifies the location of the dose points for the HI-TRAC VW transfer cask. Dose Point Locations #1 and #3 are situated below and above the water jacket, respectively. Dose Point #4 is the dose location on the HI-TRAC VW lid and dose rates below the HI-TRAC VW are estimated with Dose Point #5. Dose Point Location #2 is situated on the side of the cask at the axial mid-height.

The total dose rates presented are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Subsection 5.2.4.

Tables 5.1.1 and 5.1.2 provides dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37 and MPC-89. The dose rates listed in Table 5.1.1 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water.

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

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 5.1.2 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. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

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, adjacent, and one meter 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.

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 dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.2) are provided in Table 5.1.4 (MPC-37) for the HI-TRAC VW at a distance of 1 meter and at a distance of 100 meters. The normal condition dose rates are provided for reference. Based on the dose rate at 100 meters in Table 5.1.4, it would take 2500 hours (approximately 104 days) for the dose at the controlled area boundary to reach 5 Rem. Assuming an accident duration of 30 days, the accumulated dose at the controlled area boundary would be 1.4 Rem. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

Analyses summarized in this section demonstrate that the HI-STORM FW system, including the HI-TRAC VW transfer cask, is in compliance with the 10CFR72.106 limits.

Table 5.1.1

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-37 DESIGN BASIS FUEL
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	630	24	757	66	1477	1477
2	1862	74	<1	151	2088	2088
3	13	5	318	6	342	532
4	68	1	496	220	785	1078
5	635	2	1942	1003	3582	3582
ONE METER FROM THE HI-TRAC VW						
1	442	11	92	29	575	576
2	880	22	9	56	967	970
3	163	6	133	9	311	387
4	55	<1	276	72	404	567
5	324	1	1057	291	1673	1673

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- 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.

Table 5.1.2

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-89 DESIGN BASIS FUEL
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	230	18	2247	40	2535
2	2328	107	<1	219	2655
3	3	3	581	4	591
4	24	<1	505	138	668
5	126	2	2135	720	2983
ONE METER FROM THE HI-TRAC VW					
1	388	13	291	29	721
2	1076	30	21	74	1201
3	113	5	280	8	406
4	16	<1	300	43	360
5	75	<1	1202	202	1480

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- 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.

Table 5.1.3

DOSE RATES FOR ARRAYS OF HI-STORM FWs with MPC-37

Array Configuration	1 cask	2x2	2x3	2x4	2x5
HI-STORM FW Overpack					
45,000 MWD/MTU AND 5-YEAR COOLING					
Annual Dose (mrem/year)	13	11	16	21	10
Distance to Controlled Area Boundary (meters)	300	400	400	400	500

Notes:

- Values are rounded to nearest integer.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.

Table 5.1.4

DOSE RATES FROM HI-TRAC VW WITH MPC-37
FOR ACCIDENT CONDITIONS
AT BOUNDING BURNUP AND COOLING TIMES

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)
ONE METER FROM HI-TRAC						
HI-TRAC VW						
65,000 MWD/MTU AND 10-YEAR COOLING						
2 (Accident Condition)	942	3	10	2458	3413	3418
2 (Normal Condition)	443	46	5	113	607	609
100 METERS FROM HI-TRAC						
HI-TRAC VW						
65,000 MWD/MTU AND 10-YEAR COOLING						
2 (Accident Condition)	0.4	0.1	0.1	1.3	1.9	2

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.

Table 5.1.5

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	198	2	13	4	217	217
2	89	1	<1	1	92	92
3 (surface)	8	<1	24	2	35	49
3 (overpack edge)	8	<1	59	<1	69	104
4 (center)	0.1	0.5	0.4	0.1	1.1	1.4
4 (mid)	1	1	4	1	7	9
4 (outer)	7	<1	28	<1	37	54

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- 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.

Table 5.1.6

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	163	2	31	3	199
2	86	2	<1	1	90
3 (surface)	3	<1	29	2	35
3 (overpack edge)	5	<1	69	<1	76
4 (center)	0.1	0.4	0.4	0.1	1
4 (mid)	0.2	0.5	4.3	0.5	6
4 (outer)	2	<1	33	<1	37

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- 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.

Table 5.1.7

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	38	<1	4	<1	44	44
2	49	<1	<1	<1	52	52
3	4	<1	4	<1	10	13
4 (center)	0.4	0.3	1	0.2	1.9	2.5

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.

Table 5.1.8

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	37	<1	7	<1	46
2	45	<1	<1	<1	48
3	3	<1	5	<1	10
4 (center)	0.2	0.2	1	0.1	2

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.

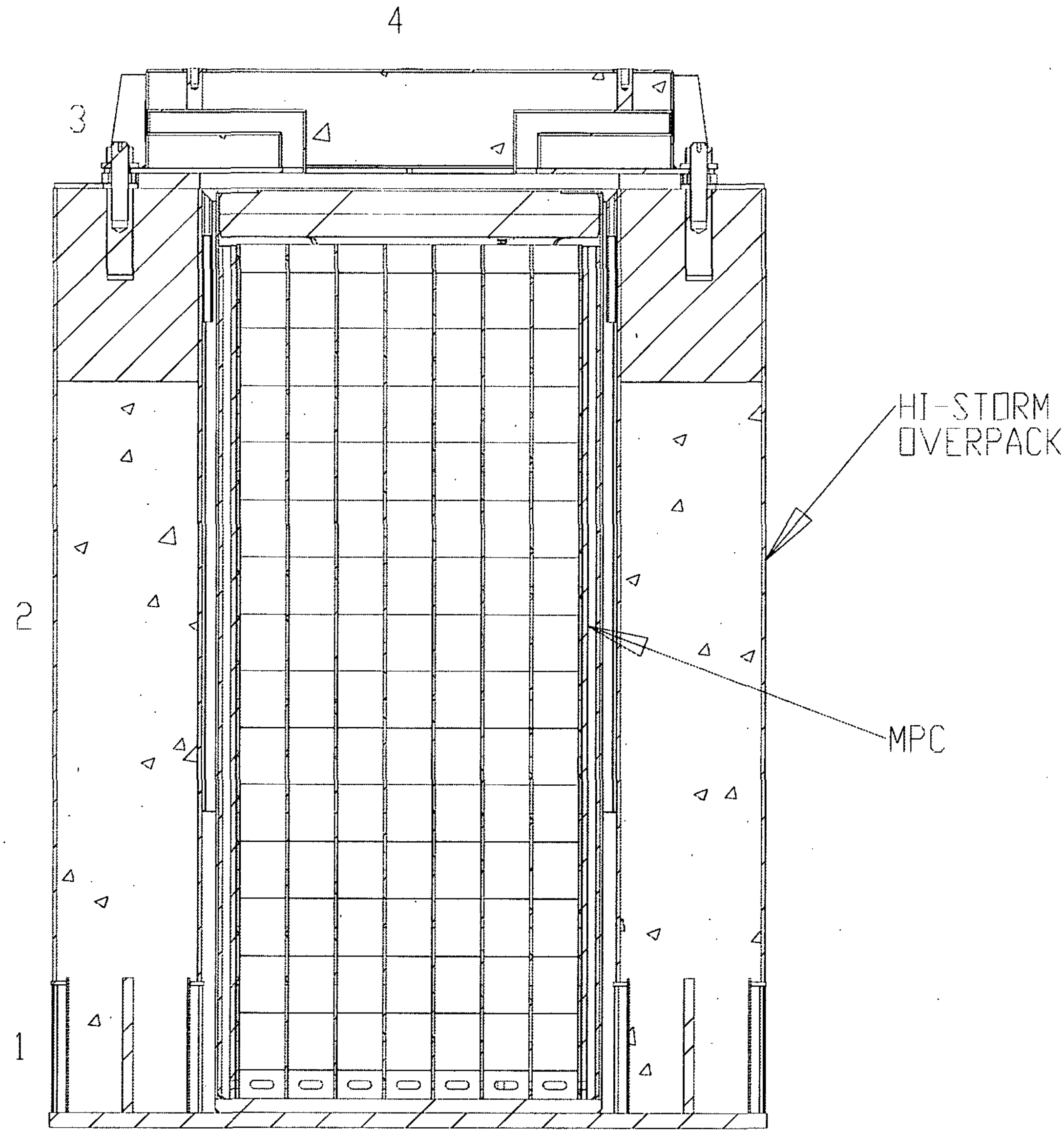

Figure 5.1.1

CROSS SECTION ELEVATION VIEW OF HI-STORM FW OVERPACK WITH DOSE POINT LOCATIONS

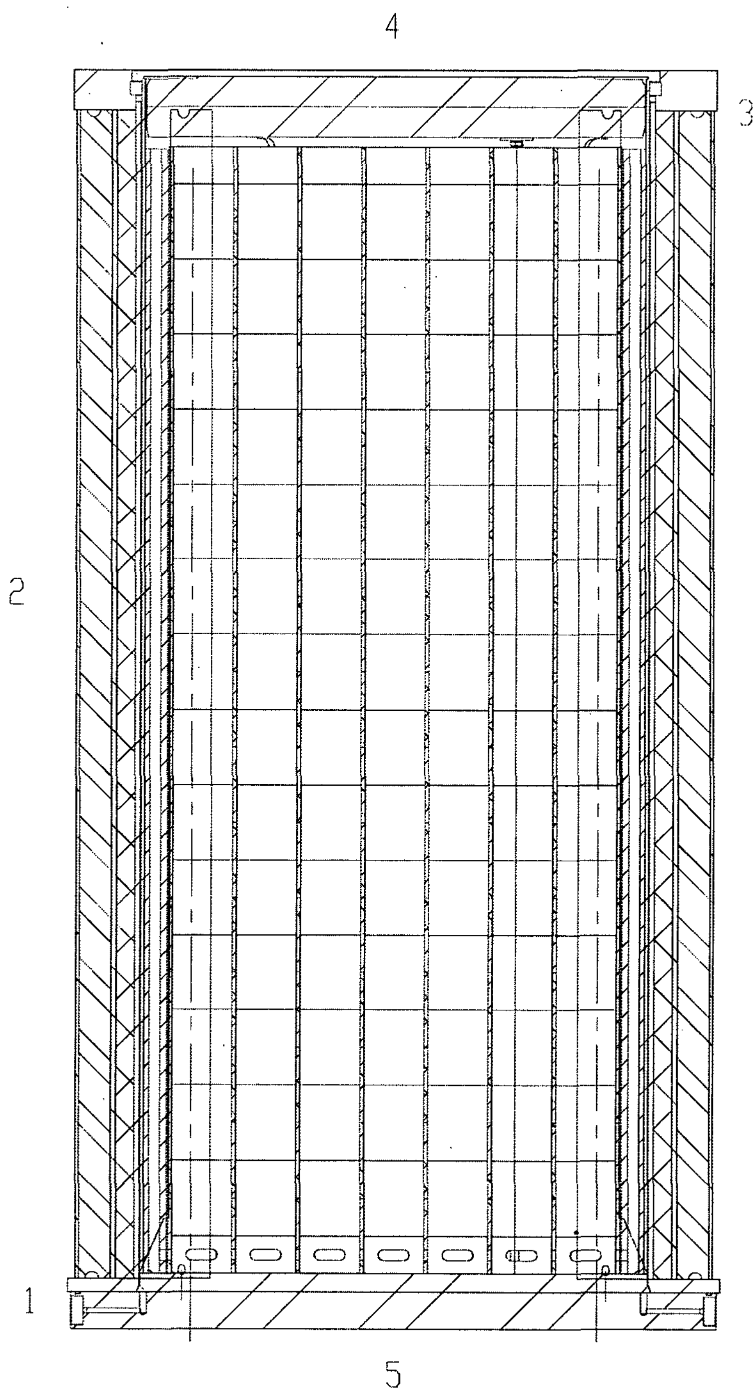

Figure 5.1.2
CROSS SECTION ELEVATION VIEW OF HI-TRAC VW TRANSFER CASK WITH DOSE POINT LOCATIONS

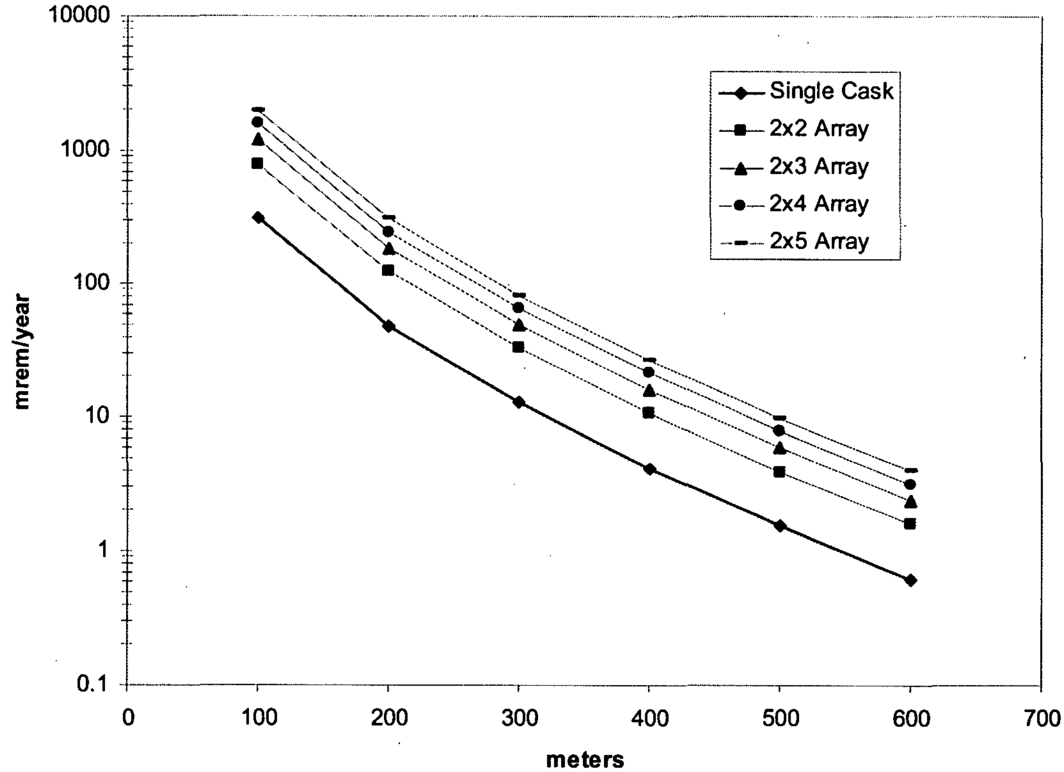

Figure 5.1.3

ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-37 FOR 45,000 MWD/MTU AND 5 YEAR COOLING
(8760 HOUR OCCUPANCY ASSUMED)

5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 5 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

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

A description of the design basis fuel for the source term calculations is provided in Table 5.2.1. Subsection 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

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

5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuel at the burnups and cooling times used for normal and accident conditions.

Previous analyses were performed for the HI-STORM 100 system to determine the dose contribution from gammas as a function of energy [5.2.17]. The results of these analyses have revealed that, due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low. Therefore, all photons with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations.

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

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

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 0.8 gm/kg was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses for an 8x8 fuel assembly were used. These masses are also appropriate for the 10x10 assembly since the masses of the non-fuel hardware from a 10x10 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation.

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

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

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.6. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.7 through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations for normal and accident conditions for the non-fuel regions of the assemblies in the MPC-37 and the MPC-89.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in

MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments of 3.2 and 3.6 wt% were chosen for the BWR and PWR design basis fuel assemblies under normal conditions, respectively. For the accident conditions, a fuel enrichment of 4.8 wt% was chosen to accommodate the higher burnups of the selected source terms (see Table 5.0.2) in accordance with Table 5.2.24 of reference [5.2.17].

The neutron source calculated for the design basis fuel assemblies for the MPCs and the design basis fuel are listed in Tables 5.2.11 through 5.2.14 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations for normal and accident conditions. The neutron spectrum is generated in ORIGEN-S.

5.2.3 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM FW system as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Subsection 2.1.

5.2.3.1 BPRAs and TPDs

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

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

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

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis W 17x17 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.6 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time.

Since the HI-STORM FW cask system is designed to store many varieties of PWR fuel, a representative TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished in the HI-STORM 100 FSAR [5.2.17] by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The TPD was determined to be the Westinghouse 17x17 guide tube plug and the BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a single hypothetical BPRA. The masses of these devices are listed in Table 5.2.15.

Table 5.2.16 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). A burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. Table 2.1.25 of the HI-STORM 100 [5.2.17] lists the allowable burnups and cooling times for non-fuel hardware that corresponds to the BPRA. These burnup and cooling times assure that the Co-60 activity remains below the levels specified above. For specific site boundary evaluations, these levels/values can be used if they are bounding. Alternatively, more realistic values can be used. Specifically, if the burnups are higher then new values should be calculated.

The HI-STORM 100 [5.2.17] presents dose rates for both BPRAs and TPDs. The results indicate that BPRAs are bounding, therefore all dose rates in this chapter will contain a BPRA in every PWR fuel location. However, Section 5.4 also contains a quantitative dose rates comparison from BPRAs and TPDs to validate this approach. Subsection 5.4.4 discusses the increase in the cask dose rates due to the insertion of BPRAs into fuel assemblies.

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.

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 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 2.1.1).

Table 5.2.1

DESCRIPTION OF DESIGN BASIS CLAD FUEL

	PWR	BWR
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	2	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	**BWR**
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 5.2.2 CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	**Upper Energy**	**45,000 MWD/MTU 5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.95E+15	3.40E+15
0.7	1.0	6.52E+14	7.67E+14
1.0	1.5	1.52E+14	1.22E+14
1.5	2.0	1.19E+13	6.79E+12
2.0	2.5	6.64E+12	2.95E+12
2.5	3.0	2.88E+11	1.05E+11
Total		2.78E+15	4.30E+15

Table 5.2.3 CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	**Upper Energy**	**65,000 MWD/MTU 10-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.79E+15	3.12E+15
0.7	1.0	2.38E+14	2.80E+14
1.0	1.5	9.79E+13	7.83E+13
1.5	2.0	7.00E+12	4.00E+12
2.0	2.5	1.46E+11	6.50E+10
2.5	3.0	1.24E+10	4.50E+09
Total		2.13E+15	3.48E+15

Table 5.2.4 CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	**Upper Energy**	**45,000 MWD/MTU 5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	7.52E+14	1.31E+15
0.7	1.0	2.40E+14	2.82E+14
1.0	1.5	5.53E+13	4.42E+13
1.5	2.0	4.15E+12	2.37E+12
2.0	2.5	2.02E+12	8.97E+11
2.5	3.0	9.74E+10	3.54E+10
Total		1.05E+15	2.04E+15

Table 5.2.5 CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	**Upper Energy**	**65,000 MWD/MTU 10-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	6.98E+14	1.21E+15
0.7	1.0	8.37E+13	9.85E+13
1.0	1.5	3.50E+13	2.80E+13
1.5	2.0	2.52E+12	1.44E+12
2.0	2.5	4.49E+10	2.00E+10
2.5	3.0	3.90E+09	1.42E+09
Total		8.19E+14	1.34E+15

Table 5.2.6

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

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

Table 5.2.7

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	45,000 MWD/MTU and 5-Year Cooling (curies)
Lower End Fitting	80.53
Gas Plenum Springs	15.70
Gas Plenum Spacer	11.15
Expansion Springs	N/A
Incore Grid Spacers	334.42
Upper End Fitting	53.57
Handle	N/A

Table 5.2.8

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	65,000 MWD/MTU and 10-Year Cooling (curies)
Lower End Fitting	49.87
Gas Plenum Springs	9.72
Gas Plenum Spacer	6.91
Expansion Springs	N/A
Incore Grid Spacers	207.09
Upper End Fitting	33.18
Handle	N/A

Table 5.2.9

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	**45,000 MWD/MTU and 5-Year Cooling (curies)**
Lower End Fitting	158.66
Gas Plenum Springs	48.48
Gas Plenum Spacer	N/A
Expansion Springs	8.81
Grid Spacer Springs	72.72
Upper End Fitting	44.07
Handle	5.51

Table 5.2.10

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	**65,000 MWD/MTU and 10-Year Cooling (curies)**
Lower End Fitting	90.17
Gas Plenum Springs	27.55
Gas Plenum Spacer	N/A
Expansion Springs	5.01
Grid Spacer Springs	41.33
Upper End Fitting	25.05
Handle	3.13

Table 5.2.11 CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY FOR 45,000 MWD/MTU BURNUP AND 5 YEAR COOLING		
Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.99E+07
4.0e-01	9.0e-01	6.52E+07
9.0e-01	1.4	6.51E+07
1.4	1.85	5.20E+07
1.85	3.0	9.69E+07
3.0	6.43	8.80E+07
6.43	20.0	8.40E+06
Totals		4.06E+08

Table 5.2.12 CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY FOR 65,000 MWD/MTU BURNUP AND 10 YEAR COOLING		
Lower Energy (MeV)	**Upper Energy (MeV)**	**65,000 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	6.30E+07
4.0e-01	9.0e-01	1.37E+08
9.0e-01	1.4	1.37E+08
1.4	1.85	1.09E+08
1.85	3.0	2.03E+08
3.0	6.43	1.84E+08
6.43	20.0	1.75E+07
Totals		8.50E+08

Table 5.2.13

CALCULATED MPC-89 BWR NEUTRON SOURCE
PER ASSEMBLY
FOR DESIGN BASIS FUEL
FOR 45,000 MWD/MTU BURNUP AND 5 YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	1.37E+07
4.0e-01	9.0e-01	2.99E+07
9.0e-01	1.4	2.99E+07
1.4	1.85	2.38E+07
1.85	3.0	4.44E+07
3.0	6.43	4.03E+07
6.43	20.0	3.86E+06
Totals		1.86E+08

Table 5.2.14

CALCULATED MPC-89 BWR NEUTRON SOURCE
PER ASSEMBLY
FOR DESIGN BASIS FUEL
FOR 65,000 MWD/MTU BURNUP AND 10 YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**65,000 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.40E+07
4.0e-01	9.0e-01	5.22E+07
9.0e-01	1.4	5.20E+07
1.4	1.85	4.15E+07
1.85	3.0	7.71E+07
3.0	6.43	7.00E+07
6.43	20.0	6.68E+06
Totals		3.24E+08

Table 5.2.15 DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE		
Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.16 DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES		
Region	**BPRA**	**TPD**
Upper End Fitting (curies Co-60)	32.7	25.21
Gas Plenum Spacer (curies Co-60)	5.0	9.04
Gas Plenum Springs (curies Co-60)	8.9	15.75
In-core (curies Co-60)	848.4	N/A

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM FW system was performed with MCNP5 [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM FW system, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.A.

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

5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM FW system, including the HI-TRAC transfer cask. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 and 5.3.2, as well as Figures 5.3.12 and 5.3.13, show cross sectional views of the HI-STORM FW overpack, MPCs, and basket cells as they are modeled in MCNP. Figures 5.3.1 and 5.3.2 were created in VISED and are drawn to scale. The inlet and outlet vents were modeled explicitly; therefore streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figures 5.3.3 and 5.3.4 show a cross sectional view of the HI-TRAC VW with the MPC-37 and MPC-89, respectively, as it was modeled in MCNP. These figures were created in VISED and are drawn to scale.

Figure 5.3.5 shows a cross sectional view of the HI-STORM FW overpack with the as-modeled thickness of the various materials.

Figure 5.3.6 shows the axial representation of the HI-STORM FW overpack with the various as-modeled dimensions indicated.

Figure 5.3.7 shows axial cross-sectional views of the HI-TRAC VW transfer casks with the as-modeled dimensions and materials specified. Figures 5.3.8 and 5.3.9 shows fully labeled radial cross-sectional view of the HI-TRAC VW transfer casks and each of the MPCs.

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 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.

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

5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.10 and 5.3.11 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM FW system. The axial locations of the basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. Further, the top lid is properly modeled with its reduced diameter, which accounts for higher localized dose rates on the top surface of the HI-STORM.

The MCNP model of the HI-TRAC transfer cask accounts for the fins through the HI-TRAC water jacket, as discussed in Subsection 5.4.1, as well as the open annulus.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM FW system and HI-TRAC shielding analyses are given in Table 5.3.2. All of the materials and their actual geometries are represented in the MCNP model.

The concrete density shown in Table 5.3.2 is the minimum concrete density analyzed in this chapter. The HI-STORM FW overpacks are designed in such a way that the concrete density in the body of the overpack can be increased to approximately 3.2 g/cm^3 (200 lb/cu-ft). Increasing the density beyond the value in Table 5.3.2 would result in a significant reduction in the dose rates. This may be beneficial based on on-site and off-site ALARA considerations.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask may be equipped with a water jacket to provide radial neutron shielding. Demineralized water (borated water) will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed for the HI-STORM 100 system [5.2.17] to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

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

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Subsection 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	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					
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

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm³)	Elements	Mass Fraction (%)
Metamic-HT[†]	2.61 (9% B_4C)	Withheld in Accordance with 10 CFR 2.390	
SS304	7.94	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	1.0
		Fe	99.0
Zircaloy	6.55	Zr	98.24
		Sn	1.45
		Fe	0.21
		Cr	0.10

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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
BWR Fuel Region Mixture	4.781 (5.0 wt% U-235)	^{235}U	3.207
		^{238}U	60.935
		O	8.623
		Zr	26.752
		N	0.014
		Cr	0.027
		Fe	0.034
		Sn	0.409
PWR Fuel Region Mixture	3.769 (5.0 wt% U-235)	^{235}U	3.709
		^{238}U	70.474
		O	9.972
		Zr	15.565
		Cr	0.016
		Fe	0.033
		Sn	0.230

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.849	SS304	100
Gas Plenum Springs (PWR)	0.23626	SS304	100
Gas Plenum Spacer (PWR)	0.33559	SS304	100
Upper End Fitting (PWR)	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
Upper End Fitting (BWR)	1.4049	SS304	100
Handle (BWR)	0.26391	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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Water w/ 2000 ppm	0.958	B-10	0.036
		B-11	0.164
		H	11.17
		O	88.63
Concrete	2.3	H	1.0
		O	53.2
		Si	33.7
		Al	3.4
		Na	2.9
		Ca	4.4
		Fe	1.4

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.1
HI-STORM FW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW
AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.2

HI-STORM FW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.3

HI-TRAC VW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW
AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.4

HI-TRAC VW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW
AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.5
CROSS SECTION OF HI-STORM FW OVERPACK

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.6
HI-STORM FW OVERPACK CROSS SECTIONAL ELEVATION VIEW

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.7
HI-TRAC VW TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW (AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.8

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-37
(AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.9

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-89
(AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.10

AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK

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Figure 5.3.11

AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK

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Figure 5.3.12

CROSS SECTIONAL VIEW OF AN MPC-37 BASKET CELL AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.13

CROSS SECTIONAL VIEW OF AN MPC-89 BASKET CELL AS MODELED IN MCNP

5.4 SHIELDING EVALUATION

The MCNP-5 code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.5 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively and have previously been utilized in the HI-STORM FSAR [5.2.17]. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.5 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.5 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel, respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

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

The dose rates at the various locations were calculated with MCNP using a two-step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in each basket region for each axial and azimuthal dose location. The second step is to multiply the dose rate per starting particle for each energy group and basket location (i.e., tally output/quantity) by the source strength (i.e. particles/sec) in that group and sum the resulting dose rates for all groups and basket locations in each dose location. The normalization of these results and calculation of the total dose rate from neutrons, fuel gammas or Co-60 gammas is performed with the following equation.

$$T_{final} = \sum_{j=1}^{M}\left[\sum_{i=1}^{N}\frac{T_{i,j}}{Fm_i} * F_{i,j}\right] \quad \text{(Equation 5.4.1)}$$

where,

T_{final} = Final dose rate (rem/h) from neutrons, fuel gammas, or Co-60

N = Number of groups (neutrons, fuel gammas) or Number of axial sections (Co-60 gammas)

M = Number of regions in the basket

$T_{i,j}$ = Tally quantity from particles originating in MCNP in group/section i and region j (rem/h)(particles/sec)

$F_{i,j}$ = Fuel Assembly source strength in group i and region j (particles/sec)

Fm_i = Source fraction used in MCNP for group i

Note that dividing by Fm_i (normalization) is necessary to account for the number of MCNP particles that actually start in group i. Also note that T_i is already multiplied by a dose conversion factor in MCNP.

The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location. The estimated variance of the total dose rate, S^2_{total}, is the sum of the estimated variances of the individual dose rates S^2_i. The estimated total dose rate, estimated variance, and relative error [5.1.1] are derived according to Equations 5.4.2 through 5.4.5.

$$R_i = \frac{\sqrt{S_i^2}}{T_i} \quad \text{(Equation 5.4.2)}$$

$$S_{Total}^2 = \sum_{i=1}^{n} S_i^2 \quad \text{(Equation 5.4.3)}$$

$$T_{Total} = \sum_{i=1}^{n} T_i \qquad \text{(Equation 5.4.4)}$$

$$R_{Total} = \frac{\sqrt{S^2_{Total}}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} (R_i \times T_i)^2}}{T_{Total}} \qquad \text{(Equation 5.4.5)}$$

where,

i	=	tally component index
n	=	total number of components
T_{Total}	=	total estimated tally
T_i	=	tally i component
S^2_{Total}	=	total estimated variance
S^2_i	=	variance of the i component
R_i	=	relative error of the i component
R_{Total}	=	total estimated relative error

Note that the two-step approach outlined above allows the accurate consideration of the neutron and gamma source spectrum, and the location of the individual assemblies, since the tallies are calculated in MCNP as a function of the starting energy group and the assembly location, and then in the second step multiplied with the source strength in each group in each location. It is therefore equivalent to a one-step calculation where source terms are directly specified in the MCNP input files, except for the following approximations:

The first approximation is that fuel is modeled as fresh UO_2 fuel (rather than spent fuel) in MCNP, with an upper bound enrichment. The second approximation is related to the axial burnup profile. The profile is modeled by assigning a source probability to each of the 10 axial sections of the active region, based on a representative axial burnup profile [5.2.17]. For fuel gammas, the probability is proportional to the burnup, since the gamma source strength changes essentially linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the neutron source strength is proportional to the burnup raised to about that power [5.4.7]. This is a standard approach that has been previously used in the licensing calculations for the HI-STAR 100 cask [5.4.8] and HI-STORM 100 system [5.2.17].

Tables 5.1.6 and 5.1.7 provide the design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1. Table 5.1.8 provides the design basis dose rates at one meter from the overpack containing the MPC-37. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Subsections 5.1.1 and 5.1.2.

Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with the water jacket filled with water (condition in which welding operations are performed). For the conditions involving a fully flooded MPC-37, the internal water level was 5 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.1 (dry MPC-37 and HI-TRAC water jacket filled with water) indicates that the dose rates in the upper and lower portions of the HI-TRAC are significantly reduced with water in the MPC.

Table 5.4.4 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with an empty water-jacket. Table 5.4.5 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with the water jacket filled with water. These results demonstrate that the dose rates on contact at the top and bottom of the HI-TRAC VW are somewhat higher in the MPC-89 case than in the MPC-37 case. However, the MPC-37 produces higher dose rates than the MPC-89 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-37 is used for the exposure calculations in Chapter 11 of the SAR.

The calculations presented herein are using a uniform loading pattern. All MPCs, however, also offer a regionalized loading pattern, as mentioned in Section 5.0 and described in Subsection 2.1. This loading pattern authorizes fuel of higher decay heat (i.e., higher burnups and shorter cooling times) to be stored in certain regions of the basket. From a shielding perspective, placing the older fuel on the outside provides shielding for the inner fuel in the radial direction. Evaluations have been performed for the HI-STORM 100 [5.2.17] where analysis of the MPC-32 and MPC-68 using the same burnup and cooling times in region 1 (which contains 38% of total number of assemblies for the MPC-32 and 47% for the MPC-68) and 2. The evaluations show that approximately 21% and 27% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. Further, approximately 1% and 2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. These results clearly indicate that the outer fuel assemblies shield almost the entire gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. Using a uniform loading pattern, rather than employing the regionalized loading scheme, in these HI-STORM FW calculations is therefore acceptable as it produces conservative dose rate values on the radial surfaces.

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

5.4.1 Streaming Through Radial Steel Fins

The HI-STORM FW overpack and the HI-TRAC VW cask utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose.

Analysis of the steel fins in the HI-TRAC has previously been performed in the HI-STORM 100 FSAR [5.2.17] and indicates that neutron streaming is noticeable at the surface of the cask. The neutron dose rate on the surface of the steel fin is somewhat higher than the circumferential average dose rate at that location. The gamma dose rate, however, is slightly lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins compared to the other areas of the water jackets.

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

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

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

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was previously performed for the HI-STORM 100 [5.2.17] to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Since the 100-ton HI-TRAC and the HI-TRAC VW are similar in design, the conclusions from the 100-ton HI-RAC evaluations are also applicable to the HI-TRAC VW.

The scenario analyzed to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse in the HI-STORM 100 [5.2.17] feature fuel debris or a damaged fuel assembly that has collapsed (which can have a higher average fuel density than an intact fuel assembly). If the damaged fuel assembly would fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 and MPC-68 with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

The results for the MPC-24 and MPC-68 calculations [5.2.17] show that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter. In addition, since the dose rate change is not significant for the 100-ton HI-TRAC, the dose rate change will not be significant for the HI-TRAC VW or the HI-STORM FW overpacks.

5.4.3 Site Boundary Evaluation

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

The methodology of calculating the dose from a single HI-STORM overpack loaded with an MPC and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters is described in the HI-STORM 100 FSAR [5.2.17]. A back row factor of 0.20 was calculated in [5.2.17], and utilized herein to calculate dose value C below, based on the results that the dose from the side of the back row of casks is approximately 16 % of the total dose.

The annual dose, assuming 100% occupancy (8760 hours), at 300 meters from a single HI-STORM FW cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed.

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

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

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

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

The doses calculated in the steps above are listed in Table 5.4.7. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM FW overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

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

5.4.4 Non-Fuel Hardware

As discussed in Subsection 5.2.3, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM FW system. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. ITTRs, which are installed after core discharge and do not contain radioactive material, may also be stored in the assembly. BPRAs, TPDs and ITTRs are authorized for unrestricted storage in an MPC. The permissible locations of the CRAs and APSRs are shown in Figure 2.1.5.

Table 5.4.8 provides the dose rates at various locations on the surface and one meter from the HI-TRAC VW due to the BPRAs and TPDs for the MPC-37. The results in Table 5.4.8 show that the BPRAs essentially bound TPDs, except for only one location where the result for TPDs is marginally higher. All dose rates with NFH in this chapter therefore assume BPRA in every assembly. Note that, even for calculations without NFH, the dose from the active region conservatively contains the contribution of the BPRA. This mainly affects dose location 1 and 2, and results for these locations are therefore identical in most tables, and don't show the dose rate difference indicated in Table 5.4.8.

Two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs in the HI-STORM FSAR [5.2.17]. The dose rate due to CRAs and APSRs was explicitly calculated for dose locations around the HI-TRAC and results were provided for the different configurations of CRAs and APSRs, respectively, in the MPCs. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (the increase in dose rate on the radial surface due to CRAs and APSRs are virtually negligible). For the surface dose rate at the bottom, the value for the CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

While the evaluations described above are based on conservative assumptions, the conclusions can vary slightly depending on the number of CRAs and their operating conditions.

5.4.5 Effect of Uncertainties

The design basis calculations presented in this chapter are based on a range of conservative assumptions, but do not explicitly account for uncertainties in the methodologies, codes and input parameters, that is, it is assumed that the effect of uncertainties is small compared to the numerous conservatisms in the analyses. To show that this assumption is valid, calculations have previously been performed as "best estimate" calculations and with estimated uncertainties added [5.4.9]. In all scenarios considered (e.g., evaluation of conservatisms in modeling assumptions, uncertainties associated with MCNP as well as the depletion analysis (including input parameters), etc.), the total dose rates long with uncertainties are comparable to, or lower than, the corresponding values from the design basis calculations. This provides further confirmation that the design basis calculations are reasonable and conservative.

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued) FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])	
Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued) FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])		
Neutron Energy (MeV)	**Quality Factor**	**(rem/hr)†/(n/cm^2-s)**
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD

MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW						
1	453	<1	458	65	977	977
2	1426	2	<1	313	1742	1742
3	5	<1	120	2	128	200
4	10	<1	212	<1	224	351
5 (bottom lid)	316	<1	1687	77	2081	2081
ONE METER FROM THE HI-TRAC VW						
1	342	<1	57	52	452	452
2	725	<1	5	114	845	846
3	118	<1	70	19	208	248
4	5	<1	119	<1	126	197
5	177	<1	923	20	1121	1121

Notes:
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.

Table 5.4.3 DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING						
Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW						
1	284	<1	282	4	571	571
2	810	5	<1	26	842	842
3	1	<1	64	<1	67	106
4	10	<1	212	<1	224	351
5 (bottom lid)	316	<1	1687	78	2082	2082
ONE METER FROM THE HI-TRAC VW						
1	182	<1	32	4	219	219
2	414	2	3	10	429	430
3	59	<1	38	1	99	121
4	5	<1	119	<1	126	197
5	177	<1	923	19	1120	1120

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.

Table 5.4.4

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD

MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	184	<1	1513	43	1741
2	2292	3	<1	579	2875
3	<1	<1	351	2	355
4	3	<1	217	<1	222
5 (bottom lid)	38	<1	1530	5	1574
ONE METER FROM THE HI-TRAC VW					
1	364	<1	197	69	631
2	1107	<1	12	168	1288
3	109	<1	154	26	290
4	<1	<1	132	<1	135
5	19	<1	864	2	886

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.

Table 5.4.5

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD
MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
ADJACENT TO THE HI-TRAC VW					
1	97	<1	926	2	1026
2	1294	10	<1	47	1352
3	<1	<1	189	<1	192
4	3	<1	217	<1	222
5 (bottom lid)	38	<1	1530	5	1574
ONE METER FROM THE HI-TRAC VW					
1	198	1	115	5	319
2	587	3	7	16	613
3	68	<1	83	1	153
4	<1	<1	132	<1	135
5	19	<1	864	2	886

Notes:
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.

Table 5.4.6 ANNUAL DOSE AT 300 METERS FROM A SINGLE HI-STORM FW OVERPACK WITH AN MPC-37 WITH DESIGN BASIS ZIRCALOY CLAD FUEL	
Dose Component	**45,000 MWD/MTU 5-Year Cooling (mrem/yr)**
Fuel gammas	10.5
^{60}Co Gammas	2.0
Neutrons	0.2
Total	12.7

Notes:
- Gammas generated by neutron capture are included with fuel gammas.
- The Co-60 gammas include BPRAs.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS HI-STORM FW ISFSI CONFIGURATIONS
45,000 MWD/MTU AND 5-YEAR COOLING ZIRCALOY CLAD FUEL

Distance	**A Side of Overpack (mrem/yr)**	**B Top of Overpack (mrem/yr)**	**C Side of Shielded Overpack (mrem/yr)**
100 meters	278	31	56
200 meters	43	5	9
300 meters	12	1	2
400 meters	4	0.4	0.8
500 meters	1	0.2	0.3
600 meters	0.6	0.06	0.1

Notes:
- 8760 hour annual occupancy is assumed.
- Values are rounded to nearest integer where appropriate.

Table 5.4.8 DOSE RATES DUE TO BPRAs AND TPDs FROM THE HI-TRAC VW FOR NORMAL CONDITIONS		
Dose Point Location	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW		
1	8.98	0.0
2	24.58	0.0
3	190.19	165.31
4	294.60	275.53
5	14.52	0.0
ONE METER FROM THE HI-TRAC VW		
1	6.45	0.40
2	14.75	3.10
3	77.71	86.95
4	164.41	153.49
5	6.75	0.0

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. 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.

5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this SAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

The shielding-significant SSCs important to safety have been valuated in this chapter and their impact on personnel and public health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM FW system has been evaluated.

It has been shown that the design of the shielding system of the HI-STORM FW system is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel in full conformance with 10CFR72.

5.6 REFERENCES

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APPENDIX 5.A

SAMPLE INPUT FILES FOR SAS2H, ORIGEN-S, AND MCNP

Withheld in Accordance with 10 CFR 2.390

CHAPTER 6†: CRITICALITY EVALUATION

6.0 INTRODUCTION

This chapter documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel in accordance with 10CFR72.124 [6.1.2]. The evaluation shows that the maximum k_{eff} value, including all applicable biases and uncertainties is below 0.95 for all normal, off-normal and accident conditions. This demonstrates that the HI-STORM FW system meets the criticality safety requirements of 10CFR72 [6.1.2] and the Standard Review Plan for Dry Cask Storage Systems (NUREG-1536) [6.1.1].

In addition, this chapter describes the HI-STORM FW system design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to provide a sufficient basis for the evaluation of the package.

Note that the analysis methodologies and modeling assumptions are identical to those utilized in the licensing of the HI-STORM 100 system in Docket No. 72-1014 ([6.0.1], Chapter 6), except for the following:

- A newer version of the Monte Carlo code MCNP, namely MCNP5, is used, together with the corresponding cross-sections. The benchmark calculations were updated accordingly.

† 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).

Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.0.2].

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1] and 10CFR72.124 [6.1.2], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM FW system, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, the results demonstrate that the HI-STORM FW system is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM FW system when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM FW system depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC;

2. The fuel basket structure which is made entirely of the Metamic-HT neutron absorber material;

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and

4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel in the PWR fuel basket.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 12 have no adverse effect on the design parameters important to criticality safety, except for the non-mechanistic tip-over event, which could result in limited plastic deformation of the basket. However, a bounding basket deformation is already included in the criticality models for normal conditions, and thus, from the criticality safety standpoint, the off-normal and accident conditions are identical to those for normal conditions.

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 mechanisms that would cause its loss or a diminution of its effectiveness (see Chapter 8, specifically Section 8.9, and Section 10.1.6.3 for further information on the qualification and testing of the neutron absorber material). Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM FW system does not rely on the use of any of the following aids to the reduction of reactivity present in the storage system:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 90 percent of the B-10 content for the Metamic-HT fixed neutron absorber undergirded by comprehensive tests as described in Subsection 10.1.6.3.

The HI-STORM FW system consists of the HI-STORM FW storage cask, the HI-TRAC VW transfer cask and Multi-Purpose-Canisters (MPCs) for PWR and BWR fuel (see Chapter 1, Table 1.0.1). Both the HI-TRAC VW transfer cask and the HI-STORM FW storage cask accommodate the interchangeable MPC designs. The HI-STORM FW storage cask uses concrete as a shield for both gamma and neutron radiation, while the HI-TRAC VW uses lead and steel for gamma radiation and a water-filled jacket for neutron shielding. The design details can be found in the drawing packages in Section 1.5.

While the MPCs are in the HI-STORM FW cask during storage, they are internally dry (no moderator), and thus, the reactivity is very low ($k_{eff} \sim 0.6$). However, the MPCs are flooded for loading and unloading operations in the HI-TRAC VW cask, which represents the limiting case in terms of reactivity. Therefore, the majority of the analyses have been performed with the MPCs in a HI-TRAC VW cask, and only selected cases have been performed for the HI-STORM FW cask.

Confirmation of the criticality safety of the HI-STORM FW system was accomplished with the three-dimensional Monte Carlo code MCNP5 [6.1.4]. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.5].

To assess the reactivity effects due to temperature changes, CASMO-4, a two-dimensional transport theory code [6.1.6] for fuel assemblies was used. CASMO-4 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects.

Benchmark calculations were made to compare the primary code package (MCNP5) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM FW system. The most important parameters are (1) the enrichment, (2) cell spacing, (3) the ^{10}B loading of the neutron absorber panels, and (4) the soluble boron concentration in the water (for PWR fuel). The critical experiment benchmarking work is summarized in Appendix 6.A.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- No credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product.

- The fuel stack density is assumed to be at 97.5% of the theoretical density for all criticality analyses. This is a conservative value, since it corresponds to a very high pellet density of 99% or more of the theoretical density. Note that this difference between stack and pellet density is due to the necessary dishing and chamfering of the pellets.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.

- When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.

- Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water. This is conservative since studies presented in Section 6.2.1 show that all assemblies are undermoderated, and that the reduction in the amount of (borated or unborated) water within the fuel assembly always results in a reduction of the reactivity. The presence of any other structural material, which would reduce the amount of water, is therefore bounded by those studies, and neglecting this material is conservative. Additionally, the potential neutron absorption of those materials is neglected.

- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.

- Planar-averaged enrichments are assumed for BWR fuel. Analyses are presented that demonstrate that the use of planar-averaged enrichments is appropriate.

- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the

Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias and bias uncertainty, two approaches are utilized. One where the results of the benchmark calculations are used directly and one where benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000. Consistent with NUREG-1536, the larger of the combined bias and bias uncertainty of the two approaches is used.

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

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

- Regarding the position of assemblies in the basket, configurations with centered and eccentric positioning of assemblies in the fuel storage locations are considered.

- For undamaged fuel assemblies, as defined in the Glossary, all fuel rod positions are assumed to contain a fuel rod. To qualify assemblies with missing fuel rods, those 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.

- For DFCs, a large ID and small wall thickness is used. This is conservative, since it maximizes the area of the optimum moderated fuel, and minimizes the neutron absorption in the DFC wall.

The design basis criticality safety calculations are performed for a single internally flooded HI-TRAC VW transfer cask with full water reflection on all sides (limiting cases for the HI-STORM FW system), for fuel assemblies listed in Chapter 2, are conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and include the calculational bias, uncertainties, and calculational statistics. In addition, a few results for single internally dry (no moderator) HI-STORM FW storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are performed to confirm the low reactivity of the HI-STORM FW system in storage.

Note that throughout this chapter reactivity results are stated as maximum neutron multiplication factor values (k_{eff}) conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics, unless otherwise noted.

For undamaged fuel, and for each of the MPC designs under flooded conditions (HI-TRAC

VW), minimum soluble boron concentration (if applicable) and fuel assembly classes[††], Tables 6.1.1 and 6.1.2 list the bounding maximum k_{eff} value, and the associated maximum allowable enrichment. The maximum allowed enrichments and the minimum soluble boron concentrations are also cited in Subsection 2.1.

For MPCs in the HI-STORM FW under dry conditions, results are listed in Table 6.1.3 for selected assembly classes.

For MPCs loaded with a combination of undamaged and damaged fuel assemblies under flooded conditions, results are listed in Tables 6.1.4 and 6.1.5. For each of the MPC designs, the tables indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum k_{eff} value, the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration. Allowed enrichments are also cited in Subsection 2.1.

These results confirm that the maximum k_{eff} values for the HI-STORM FW system are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM FW system to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM FW System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM FW storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

For PWR fuel in the MPC-37, soluble boron in the water is credited. There is a strict administrative control on the soluble boron concentration during loading and unloading of the MPC, consisting of frequent and independent measurements (For details see Subsections 9.2.2, 9.2.3, 9.2.4, and 9.4.3 and the bases for LCO 3.3.1 in Chapter 13). An accidental loss of soluble boron is therefore not credible and hence not considered.

[††] The assembly classes for BWR and PWR fuel are defined in Section 6.2.

TABLE 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 (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	1000	0.8946	1500	0.8983
14x14B	1000	0.9121	1500	0.9172
14x14C	1000	0.9211	1500	0.9277
15x15B	1500	0.9129	2000	0.9311
15x15C	1500	0.9029	2000	0.9188
15x15D	1500	0.9223	2000	0.9421
15x15E	1500	0.9206	2000	0.9410
15x15F	1500	0.9244	2000	0.9455
15x15H	1500	0.9142	2000	0.9325
15x15I	1500	0.9155	2000	0.9362
16x16A	1000	0.9275	1500	0.9366
17x17A	1500	0.9009	2000	0.9194
17x17B	1500	0.9181	2000	0.9380
17x17C	1500	0.9222	2000	0.9424
17x17D	1500	0.9183	2000	0.9384
17x17E	1500	0.9203	2000	0.9392

† 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
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
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

TABLE 6.1.3

REPRESENTATIVE k_{eff} VALUES FOR MPC-37 AND MPC-89 IN THE HI-STORM FW OVERPACK

MPC	**Fuel Assembly Class**	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
MPC-37	17x17B	5.0	0.6076
MPC-89	10x10A	4.8	0.3986

TABLE 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-37
WITH UP TO 12 DFCs

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)	Maximum k_{eff}	Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}
All 14x14, 16x16A	1300	0.9023	1800	0.9163
All 15x15, all 17x17	1800	0.9032	2300	0.9276

† 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 DFCs

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum k_{eff}
All BWR Classes except 8x8F, 9x9E/F, 10x10F and 10x10G	4.8	0.9464
8x8F, 9x9E/F and 10x10G	4.0	0.9299
10x10F	4.6	0.9428

6.2 SPENT FUEL LOADING

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in defining the authorized contents could limit or complicate the applicability of the HI-STORM FW system. To resolve this limitation, a number of fuel assembly classes for both fuel types (PWR and BWR) are defined based on bounding fuel dimensions. The results of parametric studies justify using those bounding fuel dimensions for defining the authorized contents.

6.2.1 Definition of Assembly Classes

For each array size the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; and (3) number and locations of guide tubes (PWR) or water rods (BWR). The assembly classes for PWR and BWR fuel are defined in Chapter 2, Tables 2.1.2 and 2.1.3, respectively. It should be noted that these assembly classes are consistent with the class designations in the HI-STORM 100 FSAR (Docket No. 72-1014). Specifically, assembly classes with the same identifier refer to the same set of limiting dimensions. However, some classes have been removed and others have been added compared to the HI-STORM 100.

In HI-STORM 100 FSAR (Docket No. 72-1014), extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

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

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely 1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:

- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.

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

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

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536 (see section 6.4.2.3), a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not

by a positive increase in reactivity from the reduction in fuel (which would be counter-intuitive). Therefore, in order not to overstate the conservative effect of the flooded fuel-to-clad gap, the calculations for the variation of the fuel pellet diameter are performed for a flooded gap of constant thickness by also changing the clad ID.

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 class 10x10G has 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). 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. 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 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.

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 to 8 inches of the active fuel length. This changes the fuel to water ratio in these areas, which could have an effect on reactivity. However, the top and bottom of the active length are areas with high neutron leakage, and changes in these areas typically have no significant effect on reactivity. Studies with up to 12 inches of annular pellets at the top and bottom performed for the HI-STORM FW with various pellet IDs (see Table 6.2.6) confirm this, i.e., shown no significant reactivity effects, even if the annular region of the pellet is flooded with pure water. All calculations for PWR fuel assemblies are therefore performed with solid fuel pellets along the entire length of the active fuel region, and the results are directly applicable to those PWR assemblies with annular fuel pellets. This is consistent with the HI-STORM 100, where the same analyzed conditions are analyzed and qualified.

TABLE 6.2.1

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-37 with 2000 ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum k_{eff}	standard deviation
17x17B (5.0 wt% Enrichment)	Reference	0.9374	0.0004
increase pellet OD and clad ID (+0.004)	0.0052	0.9426	0.0003
decrease pellet OD and Clad ID (-0.004)	-0.0058	0.9316	0.0004
increase clad OD (+0.004)	-0.0014	0.9360	0.0004
decrease clad OD (-0.004)	0.0017	0.9391	0.0004
increase guide tube thickness (+0.004)	-0.0001	0.9373	0.0004
decrease guide tube thickness (-0.004)	0.0004	0.9378	0.0003
remove guide tubes (i.e., replace the guide tubes with water)	0.0009	0.9383	0.0004
reduced active length (100 Inches)	-0.0020	0.9354	0.0004
increase rod pitch (+0.004)	0.0019	0.9393	0.0004
reduce rod pitch (-0.004)	-0.0017	0.9357	0.0004

TABLE 6.2.2

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-89
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum k_{eff}	standard deviation
10x10A (Reference, full-length rods only)	Reference	0.9429	0.0004
increase pellet OD and Clad ID (+0.004)	0.0037	0.9466	0.0004
decrease pellet OD and Clad ID (-0.004)	-0.0042	0.9387	0.0004
increase clad OD (+0.004)	-0.0021	0.9408	0.0003
decrease clad OD (-0.004)	0.0032	0.9461	0.0004
increase water rod thickness (+0.004)	0.0002	0.9431	0.0004
decrease water rod thickness (-0.004)	0.0009	0.9438	0.0004
remove water rods (i.e., replace the water rod tubes with water)	0.0031	0.9460	0.0004
reduced active length (100 Inches)	-0.0026	0.9403	0.0004
remove channel	-0.0113	0.9316	0.0003
increase channel thickness (+0.020)	0.0007	0.9436	0.0003
full-length and part-length rods (real assembly)	-0.0054	0.9375	0.0004
part-length rods extended to full-length	-0.0102	0.9327	0.0004
increased rod pitch (+0.004)	0.0050	0.9479	0.0004
reduced rod pitch (-0.004)	-0.0043	0.9386	0.0003
distributed enrichment, Case 1	-0.0011	0.9418	0.0003
distributed enrichment, Case 2	+0.0004	0.9433	0.0003
distributed enrichment, Case 3	-0.0099	0.9330	0.0004
distributed enrichment, Case 4	-0.0121	0.9308	0.0003

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**
14x14A	0.8983	0.8962	-0.0021
14x14B	0.9172	0.9134	-0.0038
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
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**
7x7B	0.9317	0.9261	-0.0056
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
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

Table 6.2.4

COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND DISTRIBUTED ENRICHMENTS

Case	Planar Average Enrichment (wt%)	Peak Rod Enrichment (wt%)	Maximum k_{eff}	
			Planar Average Enrichment (wt%)	Peak Rod Enrichment (wt%)
8x8C	3.01	3.80	0.8358	0.8309
8x8C	3.934	4.9	0.8975	0.8899
8x8D	3.42	3.95	0.8628	0.8636
8x8D	3.78	4.40	0.8862	0.8855
8x8D	3.90	4.90	0.8934	0.8913
9x9B	4.34	4.71	0.9195	0.9179
9x9D	3.35	4.34	0.8575	0.8456
Hypothetical #1 (48 outer rods of 3.967%E, 14 inner rods of 5.0%)	4.20	5.00	0.9104	0.9102
Hypothetical #2 (48 outer rods of 4.354%E, 14 inner rods of 5.0%)	4.50	5.00	0.9258	0.9247

Table 6.2.5

VARIATIONS OF NUMBER OF FUEL AND/OR WATER RODS FOR ASSEMBLY CLASSES 8x8B AND 8x8D (see Appendix B)

Case	**Maximum k_{eff}**
Assembly Class 8x8B	
63 Fuel Rods (Reference)	0.9369
64 Fuel Rods	0.9342
Assembly Class 8x8D	
60 Fuel Rods, no water rods modeled (Reference)	0.9380
60 Fuel Rods, 2 larger, 2 smaller water rods	0.9362
60 Fuel Rods, 4 larger water rods	0.9347
60 Fuel Rods, 4 smaller water rods	0.9359
60 Fuel Rods, 1 large water rods	0.9343
61 Fuel Rods, 3 water rods	0.9354

VARIATION OF WATER ROD LOCATIONS FOR ASSEMBLY CLASS 9x9E/F (see Appendix B)

Case	**Maximum k_{eff}**
Adjacent Water Rods (Reference)	0.9346
Water Rods separated by a Fuel Rod	0.9313

VARIATION OF GUIDE RODS FOR ASSEMBLY CLASS 15x15I (see Appendix B)

Case	**Maximum k_{eff}**
No Guide Rods (Reference)	0.9362
8 Guide Rods	0.9313

Table 6.2.6

EFFECT OF ANNULAR PELLETS IN THE TOP AND BOTTOM 12 INCHES OF THE ACTIVE REGION

Diameter of Annulus in Pellets	**Maximum k_{eff}**
Assembly Class 17x17B, 5% Enrichment, Undamaged Fuel	
None (Reference)	0.9380
0.1 Inches	0.9382
0.2 Inches	0.9379
0.3 Inches	0.9371
0.4 Inches	0.9371
0.5 Inches	0.9363
0.6 Inches	0.9368
Assembly Class 16x16A, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9163
0.2 Inches	0.9162
Assembly Class 15x15F, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9276
0.2 Inches	0.9266

Table 6.2.7

EFFECT OF PARTIAL LENGTH RODS FOR ASSEMBLY CLASS 10x10G

Length of Partial Length Rods as a Percentage of Full Length Rods	**Maximum k_{eff}**
100%	0.9466
75%	0.9455
50%	0.9404
25%	0.9286
0%	0.9208

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1 through 6.3.5 show representative cross sections of the criticality models for the two baskets. Figures 6.3.1 and 6.3.2 show a single cell from each of the two baskets. Figures 6.3.3 and 6.3.4 show the entire MPC-37 and MPC-89 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.

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 four 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.

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 conditions is that the permanent deflection of the basket panels is limited to a fraction of 0.005 (0.5%) of the panel width (see Chapter 3). The analyses in Chapter 3 demonstrate that permanent deformations of the basket walls during accident conditions are far below this limit. In fact, the analyses show that the vast majority of the basket panels remain elastic during and after an accident, and therefore show no permanent deflection whatsoever, and that any deformation is limited to small localized areas. Nevertheless, it is conservatively assumed that 2 adjacent cell walls in each cell are deflected to the maximum extent possible over their entire length and width, i.e. that the cell ID is reduced by 0.5% of the cell width, or 0.045" for the MPC-37 cells and 0.030" for the MPC-89 cells. Stated differently, the minimum cell ID based on tolerances was further reduced by the amounts stated above for all cells in each basket to account for the potential deflections of basket walls during accident conditions. Assuming that all cell sizes are reduced is a simplifying, but very conservative assumption, since cell walls are shared between neighboring cells, so while the deflection of a basket wall would reduce the cell size on one side, it necessarily increases that on the other side of the wall. MCNP5 was used to determine the manufacturing tolerances and deflections that produced the most adverse effect on criticality. After the reactivity effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case conditions in the direction which would increase reactivity. For simplification, the same worst case conditions are used for both normal and accident conditions. For all calculations, fuel assemblies were assumed to be eccentrically located in the cells, since this results in higher reactivities (see Section 6.3.3). Maximum k_{eff} results (including the bias, uncertainties, or calculational statistics), along with the selected dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for both baskets. The cell ID is evaluated for minimum (tolerance only), minimum with deformation, nominal and an increased value. The wall thickness is evaluated for nominal and minimum values.

Based on the calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined for the basket designs. Because the reactivity effect (positive or negative) of the manufacturing tolerances is not assembly dependent, these dimensional assumptions were employed for all criticality analyses.

The basket is manufactured from individual slotted panels. The panels are expected to be in direct contact with each other (see Drawings in Chapter 1). However, to show that small gaps between panels would have essentially no effect on criticality, calculations are performed with a postulated 0.06" gap between panels, repeated in the axial direction every 10" in all panels. Since it is expected that the effect of these gaps would be small, these calculations were performed with a larger number of particles per cycle, larger number of inactive cycles, and a larger total number of cycles to improve the statistics of each run, so the real reactivity effect could be better separated from the statistical "noise". The results are summarized in Tables 6.3.6 and show that the METAMIC gap has a very small effect. Therefore, all calculations are performed without any gaps between panels.

Variations of water temperature in the cask were analyzed using CASMO-4. The analyses were performed for the assembly class 10x10A in the MPC-89, and for the assembly class 17x17B with 2000 ppm soluble boron in the water in the MPC-37. 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 he 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.

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

The potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection The calculations in this subsection serve to identify the eccentric positioning of assemblies in the fuel storage locations, which results in a higher maximum k_{eff} value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum k_{eff} value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- Cell Center Configuration: All assemblies centered in their fuel storage cell;
- Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and
- Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical phenomenon that could move all assemblies within a basket consistently to the center or periphery. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.5, results are presented for all representative conditions. The table shows the maximum k_{eff} value for centered and the two eccentric configurations for each condition, and the difference in k_{eff} between the centered and eccentric positioning. In all cases, moving the assemblies and DFCs to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position, and moving the assemblies and DFCs towards the center results in an increase in reactivity, compared to the cell centered position. All calculations are therefore performed with assemblies/DFCs moved towards the center of the basket.

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	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0008	-0.0035	
70°C	-0.0023	-0.0100	
100°C	-0.0042	-0.0180	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0036	-0.0282	
100°C	-0.0096	-0.0463	

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-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

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")

TABLE 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

UO_2, DENSITY 10.686 g/cm^3 (97.5% of theoretical density of 10.96 g/cm^3)					
Nuclide	**Wgt. Fraction, 4.0 wt%**	**Wgt. Fraction, 4.5 wt%**	**Wgt. Fraction, 4.7 wt%**	**Wgt. Fraction, 4.8 wt%**	**Wgt. Fraction, 5.0 wt%**
8016	0.1185	0.1185	0.1185	0.1185	0.1185
92235	0.03526	0.03967	0.04143	0.04231	0.04408
92238	0.84624	0.84183	0.84007	0.83919	0.83742

WATER (unborated and borated), DENSITY 1.0 g/cm^3							
Nuclide	**Wgt. Fraction, 0 ppm**	**Wgt. Fraction, 1000 ppm**	**Wgt. Fraction, 1300 ppm**	**Wgt. Fraction, 1500 ppm**	**Wgt. Fraction, 1800 ppm**	**Wgt. Fraction, 2000 ppm**	**Wgt. Fraction, 2300 ppm**
5010	0.000E+00	1.800E-04	2.340E-04	2.700E-04	3.240E-04	3.600E-04	4.140E-04
5011	0.000E+00	8.200E-04	1.066E-03	1.230E-03	1.476E-03	1.640E-03	1.886E-03
1002	0.11190	0.11179	0.11175	0.11173	0.11170	0.11167	0.11164
8016	0.88810	0.88721	0.88695	0.88677	0.88650	0.88633	0.88606

METAMIC HT, 9% B_4C, DENSITY 2.6 g/cm^3	
Nuclide	**Wgt. Fraction**
13027	0.91
6000	0.01956
5010	0.01289
5011	0.05755

TABLE 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

ZR CLAD, DENSITY 6.55 g/cm^3	
Nuclide	**Wgt. Fraction**
40000	1.0
STAINLESS STEEL, DENSITY 7.84 g/cm^3	
Nuclide	**Wgt. Fraction**
24000	0.190
25055	0.020
26000	0.695
28000	0.095
ALUMINUM, DENSITY 2.7 g/cm^3	
Nuclide	**Wgt. Fraction**
13027	1.0
CONCRETE, DENSITY 2.35 g/cm^3	
Nuclide	**Wgt. Fraction**
1001	0.006
8016	0.500
11000	0.017
13027	0.048
14000	0.315
19000	0.019
20000	0.083
26000	0.012
LEAD, DENSITY 11.34 g/cm^3	
Nuclide	**Wgt. Fraction**
82000	1.0

TABLE 6.3.5

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT (FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

CASE	Contents centered (Reference)	Content moved towards center of basket		Content moved towards basket periphery	
	Maximum k_{eff}	Maximum k_{eff}	k_{eff} Difference to Reference	Maximum k_{eff}	k_{eff} Difference to Reference
MPC-37, Undamaged Fuel	0.9327	0.9380	0.0053	0.9143	-0.0184
MPC-37, Undamaged Fuel and Damaged Fuel/Fuel Debris (12 DFCs)	0.9260	0.9276	0.0016	0.9158	-0.0102
MPC-89, Undamaged Fuel	0.9369	0.9435	0.0066	0.9211	-0.0158
MPC-89, Undamaged Fuel and Damaged Fuel/Fuel Debris (16 DFCs)	0.9415	0.9451	0.0036	0.9301	-0.0114

TABLE 6.3.6

REACTIVITY EFFECTS GAPS IN BASKET CELL PLATES

Gaps in Metamic-HT	**MPC-37 (17x17B, 5.0% ENRICHMENT)**	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
None	0.9380	0.9435
0.06" every 10"	0.9382	0.9439

TABLE 6.3.7

RADIAL AND AXIAL DIMENSIONS OF THE HI-TRAC VW IN THE MCNP MODELS

Component / Material	Thickness (Inches)
Radial Direction (Inside to Outside)	
MPC Shell (Steel)	0.5
Water between MPC and HI-TRAC VW	0.125
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Lead	2.75
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Water Jacket	4.75
HI-TRAC VW Shell (Steel)	0.5
External Water Reflector	12
Axial Direction (Bottom to Top)	
External Water Reflector	12
HI-TRAC VW Bottom Lid (Steel)	5.5
MPC Base Plate (Steel)	3
Water	2
Active Fuel Region	150
Water	6
MPC Lid (Steel)	9
External Water Reflector	12

TABLE 6.3.8

MCNP CROSS SECTION SETS USED IN THE ANALYSES

Nuclide	MCNP Cross Section Set [6.1.4]
1001	62c
5010	66c
5011	66c
6000	66c
8016	62c
13027	62c
24000	50c
25055	62c
26000	55c
28000	50c
40000	66c
82000	50c
92235	69c
92238	69c

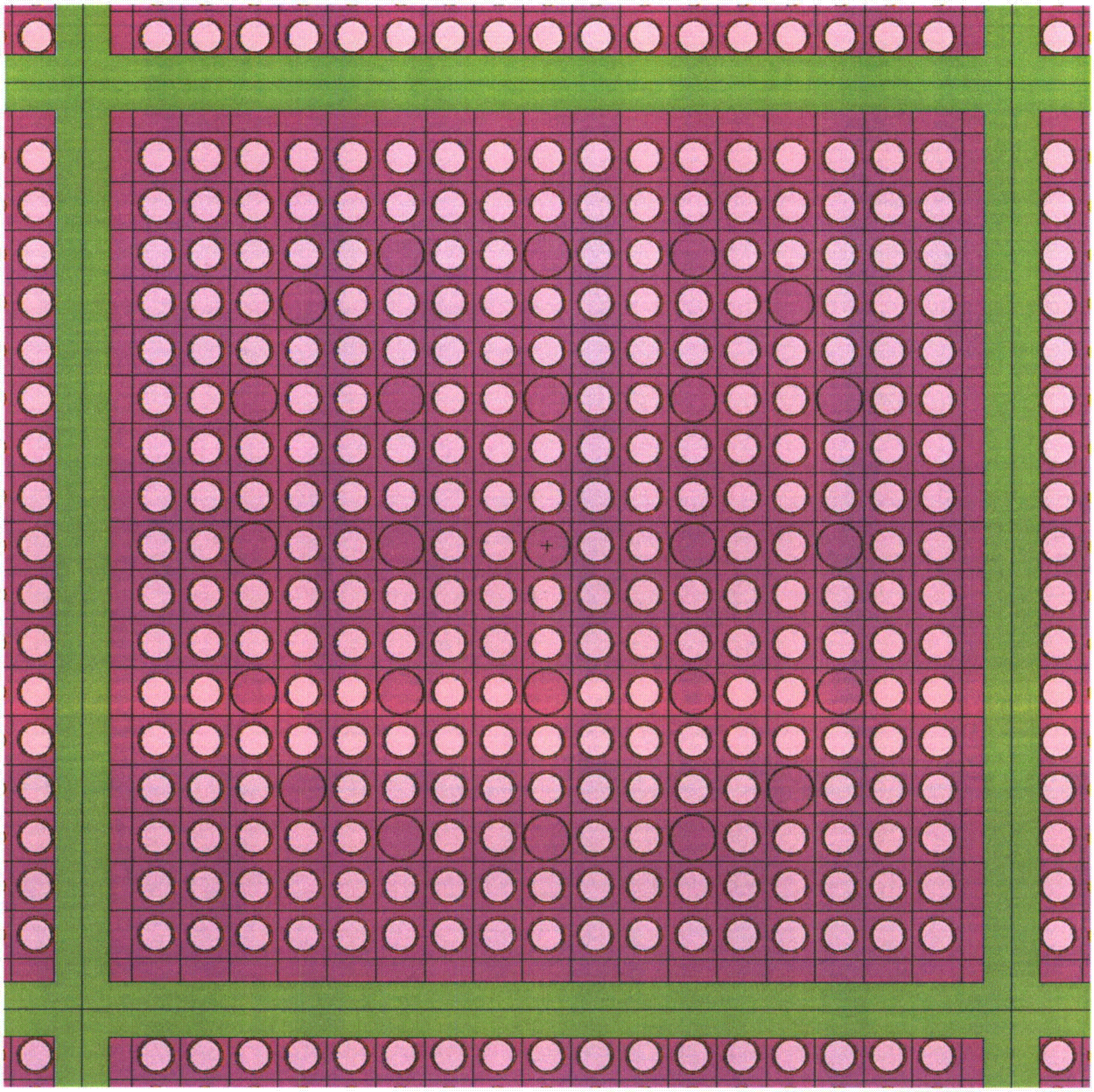

Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.1: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-37

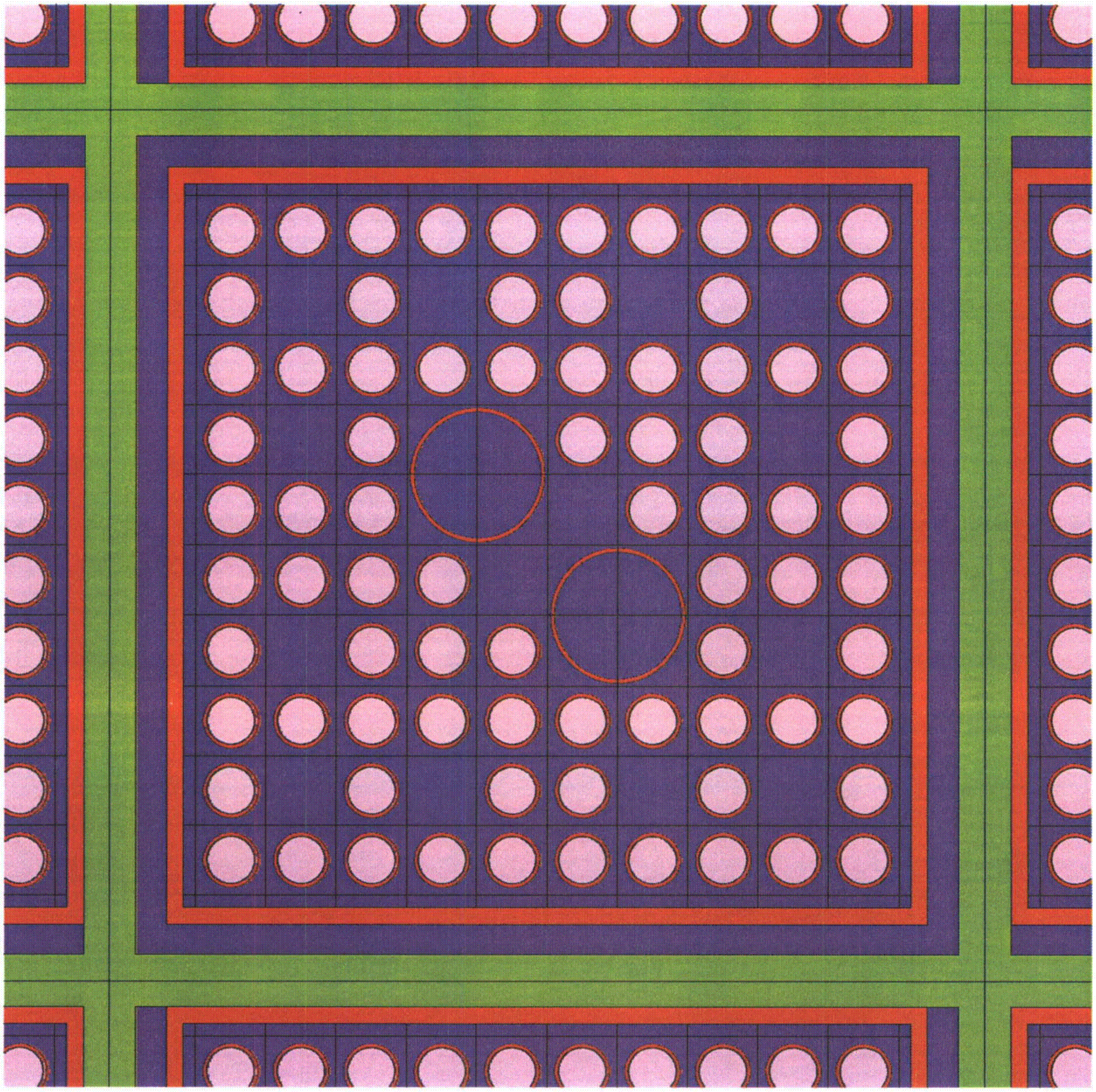

Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.2: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-89

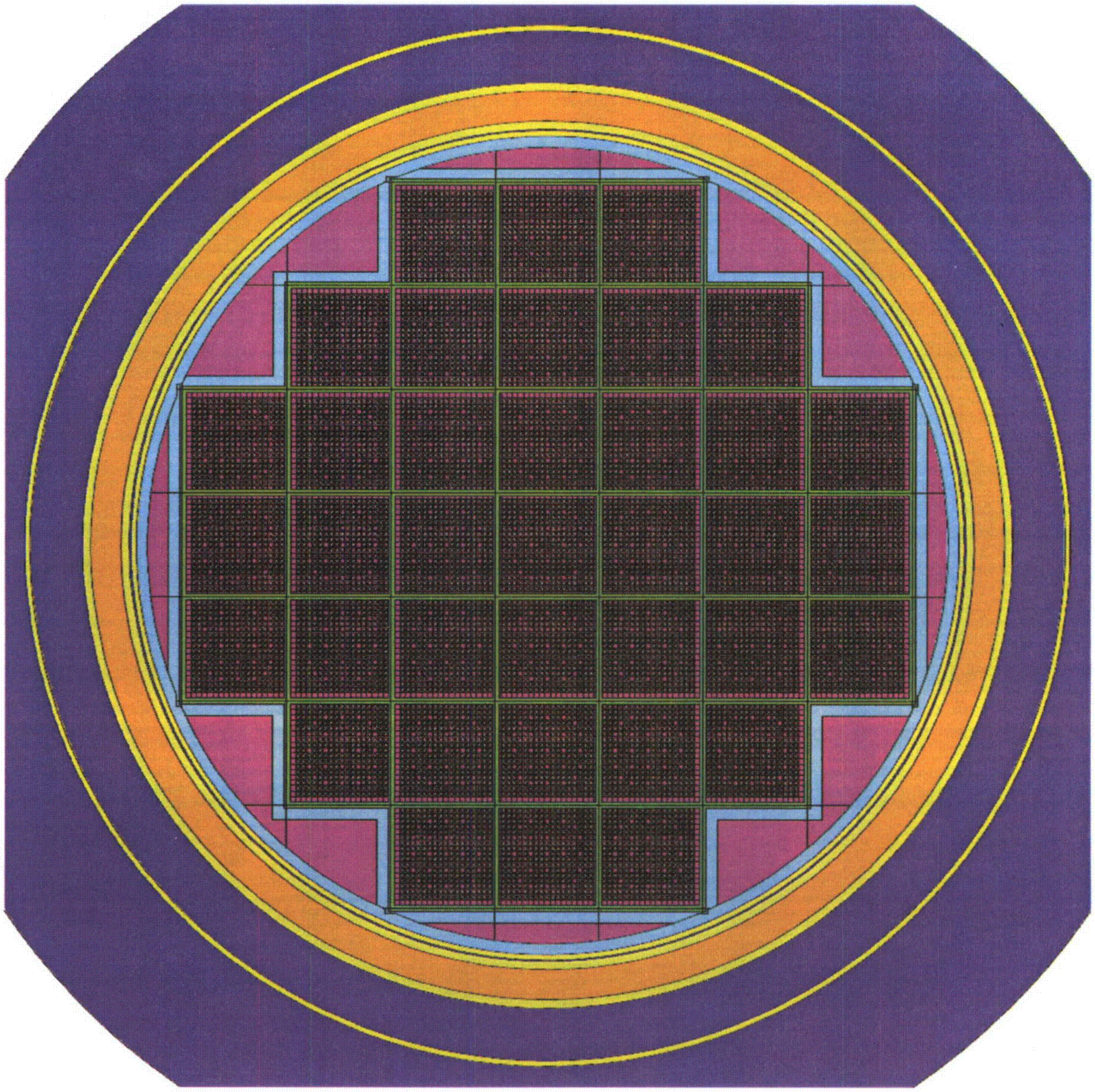

Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.3: Calculational Model (planar cross-section) of the MPC-37

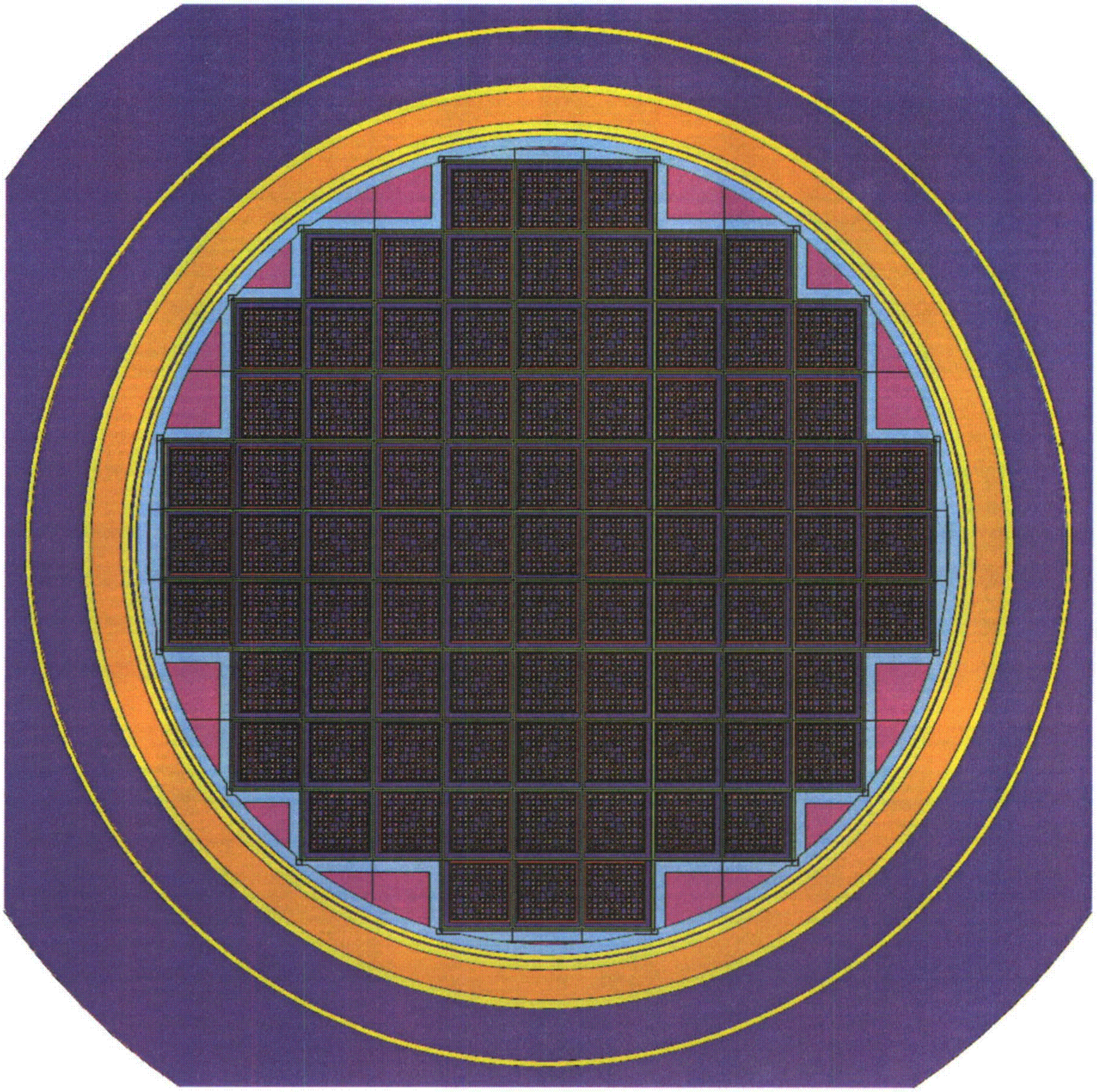

Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.4: Calculational Model (planar cross-section) of the MPC-89

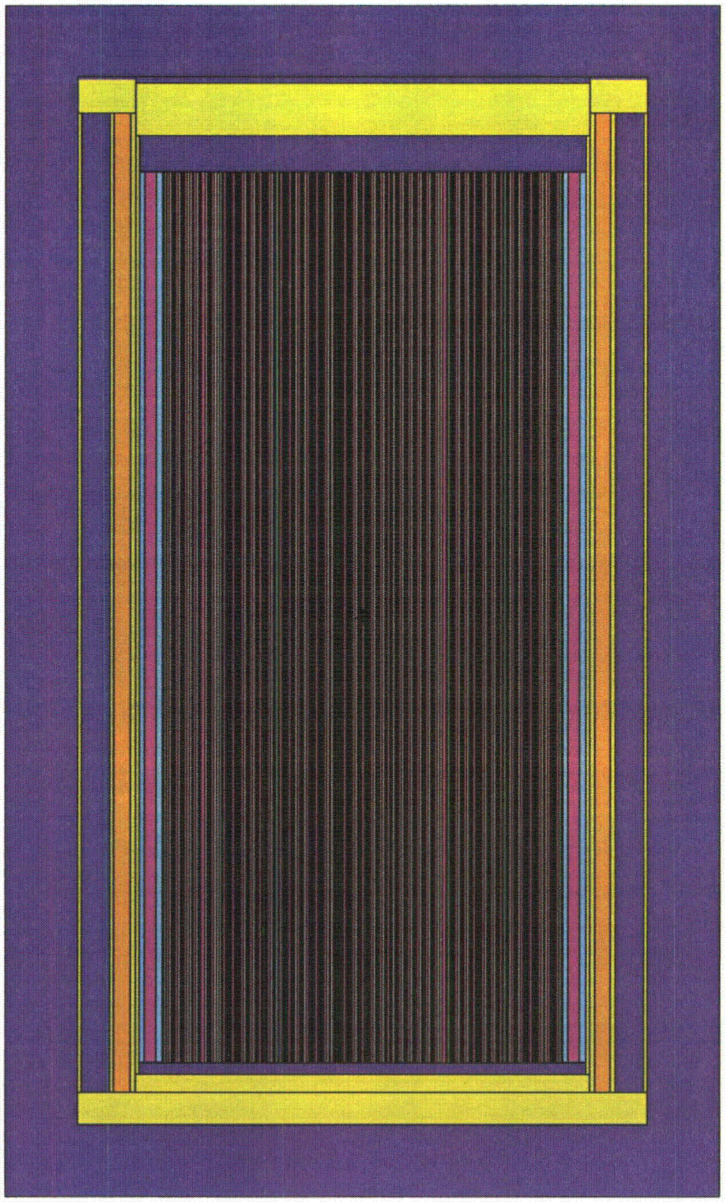

Figure generated directly from MCNP input file using the MCNP plot function. For axial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.5: Calculational Model in Axial Direction

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational Methodology

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP5 [6.1.4] developed at the Los Alamos National Laboratory. MCNP5 was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP5 calculations used continuous energy cross-section data distributed with the code [6.1.4].

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP5 criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. Based on this information, a minimum of 20,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). To verify that these parameters are sufficient, studies were performed where the number of particles per cycle and/or the number of skipped cycles were increased. The calculations are presented in Table 6.4.9, and show only small differences between the cases, with the statistical tolerance of those calculations. All calculations are therefore performed with the parameters stated above, except for some studies that are performed with 50000 neutrons per cycle for improved accuracy, and except for the calculations for the HI-STORM, which need less particles for convergence. Appendix 6.D provides sample input files for the MPC-37 and MPC-89 basket in the HI-STORM FW system.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Section 2.1. The calculations were based on the assumption that the HI-STORM FW system (HI-TRAC VW transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel assemblies and basket 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 types, 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.

Internal and External Moderation (Section 6.4.2.1): The studies presented in Table 6.2.3 show that all assemblies essentially behave identical in respect to water moderation, specifically, that all assemblies are undermoderated. The principal effect of changes to the internal and external moderation would therefore be independent of the fuel type.

Partial Flooding (Section 6.4.2.2): The partial flooding of the basket, either in horizontal or vertical direction, reduces the amount of fuel that partakes effectively in the thermal fission process, while essentially maintaining the fuel-to-water ratio in the volume that is still flooded. This will therefore result in a reduction of the reactivity of the system (similar to that of the reduction of the active length), and due to the similarity of the fuel assemblies is not dependent on the specific fuel type.

Pellet-to-clad Gap (Section 6.4.2.3): As demonstrated by the studies shown in Section 6.2.1, all assemblies are undermoderated. Flooding the pellet-to-clad gap will therefore improve the moderation and therefore increase reactivity for all assembly types.

Preferential Flooding (Section 6.4.2.4): The only preferential flooding situation that may be credible is the flooding of the bottom section of the DFCs while the rest of the MPC internal cavity is already drained. In this condition, the undamaged assemblies have a negligible effect on the system reactivity since they are not flooded with water. The dominating effect is from the damaged fuel model in the DFCs. However, the damaged fuel model is conservatively based on an optimum moderated array of bare fuel rods in water, and therefore representative of all fuel types. The results are therefore applicable to all fuel types.

6.4.2.1 Internal and External Moderation

Calculations in this section demonstrate that the HI-STORM FW system remains subcritical for all credible conditions of moderation.

6.4.2.1.1 External Moderator Density

Calculations for the MPC designs with external moderators of various densities are shown in Table 6.4.1, all performed for the HI-TRAC VW and the MPC fully flooded. The results show that the maximum k_{eff} is essentially independent from the external water density. Nevertheless, all further evaluations are performed with full external water density.

6.4.2.1.2 Internal Moderator Density

In a definitive study, Cano, et al. [6.4.1] have demonstrated that the phenomenon of a peak in reactivity at low moderator densities (sometimes called "optimum" moderation) does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations were made to confirm that the phenomenon does not occur with low density water inside the casks.

Calculations for the MPC designs with internal moderators of various densities are shown in Table 6.4.5. Results show that in all cases the reactivity reduces with reducing water density, with both filled and voided guide and instrument tubes for PWR assemblies (see Section 6.4.7). All further calculations are therefore performed with full water density inside the MPCs.

6.4.2.2 Partial Flooding

Calculations in this section address partial flooding in the HI-STORM FW system and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cm^3) water and the remainder of the cask is filled with steam consisting of ordinary water at a low partial density (0.002 g/cm^3 or less), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

6.4.2.3 Clad Gap Flooding

As recommended by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e., different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell at the bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 12). For damaged fuel assemblies and fuel debris, the assemblies or debris are loaded into stainless steel Damaged Fuel Containers fitted with mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Preferential flooding of the MPC basket is therefore not possible.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. As a simplifying and conservative approach to model this condition it is assumed that the DFCs are completely flooded while the remainder of the MPC is only filled with steam consisting of ordinary water at partial density (0.002 g/cm^3 or less). Assuming this condition, the case resulting in the highest maximum k_{eff} for the fully flooded condition (see Subsection 6.4.4) is re-analyzed assuming the preferential flooding condition. Table 6.4.4 lists the maximum k_{eff} in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{eff} than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC Confinement Boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Subsection 6.4.4.

6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 12 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC VW transfer cask and minor damage to the concrete radiation shield for the HI-STORM FW storage cask, which have no adverse effect on the design parameters important to criticality safety, and to minor deformation of the basket geometry, which is already considered in the analyses for the normal conditions.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM FW system is in full compliance with the requirement of 10CRF72.124, which states that "before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety."

6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) and summarized in Section 6.1. To demonstrate the applicability of the HI-TRAC VW analyses, results of the design basis criticality safety calculations for the HI-TRAC VW cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-4) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff}^{max} = k_c + K_c \sigma_c + Bias + \sigma_B$$

where:

- ⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;
- ⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.5]. Each final k_{eff} value is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;
- ⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code;
- ⇒ ***Bias*** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and
- ⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

6.4.4 Damaged Fuel and Fuel Debris

6.4.4.1 Generic Approach

All MPCs are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into DFCs. The number and permissible location of DFCs is provided in Table 2.1.1 and the licensing drawing in Section 1.5, respectively. Because the entire height of the fuel basket contains the neutron absorber (Metamic-HT), the DFCs are covered by the neutron absorber even if they were to move axially.

Damaged fuel assemblies are assemblies with known or suspected cladding defects greater than pinholes or hairlines, or with missing rods, but excluding fuel assemblies with gross defects (for a full definition see Glossary). Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

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 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
- 10x10A for all other BWR assemblies;
- 16x16A for all PWR assemblies with 14x14 and 16x16 arrays; and
- 15x15F for all PWR assemblies with 15x15 and 17x17 arrays.

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. 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 of the assembly class 15x15F selected as the bounding assembly for the cases with undamaged and damaged fuel. The effect of the active fuel length is less than that, with a value of 0.0026 reported in Table 6.2.1 for a much larger difference in active length of 50 Inches. The difference in active length between the 17x17D/E and 15x15F is therefore more than bounded, and the 15x15F assembly class is therefore appropriate to bound all undamaged assemblies with 15x15 and 17x17 arrays.

Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e., all cladding and other structural material in the DFC is replaced by water.

- For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.

- The active length of these rods is assumed to be the same as for the intact fuel rods in the basket, even for more densely packed bare fuel rod arrays where it results in a total amount of fuel in the DFC that exceeds that for the intact assembly.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 16 (4x4) and 324 (18x18) for BWR fuel, and between 64 (8x8) and 576 (24x24) for PWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

Further, this is a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 150 inch (BWR and PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

All calculations are performed for full cask models, containing the maximum permissible number of DFCs together with undamaged assemblies.

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 containing a 14x14 array of bare fuel rods.

Principal results are listed in Table 6.4.6 and 6.4.7 for the MPC-37 and MPC-89, respectively. The highest maximum k_{eff} values correspond to a 16x16 array of bare fuel rods for the MPC-37, and for an 11x11 array for the MPC-89. 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. 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.

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%)
100%	100%	0.9380	0.9435
100%	70%	0.9377	0.9432
100%	50%	0.9399	0.9439
100%	20%	0.9366	0.9428
100%	10%	0.9374	0.9437
100%	5%	0.9376	0.9435
100%	1%	0.9383	0.9435

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

TABLE 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	**Maximum k_{eff}**	
	MPC-37 (17x17B, 5.0% ENRICHMENT)	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
dry	0.9335	0.9391
flooded with unborated water	0.9380	0.9435

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

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 %**	
Guide Tubes	N/A	filled	void	filled	void	filled	void
1.00	0.9435	0.9181	0.9071	0.9380	0.9292	0.9276	0.9265
0.99	0.9415	0.9181	0.9059	0.9367	0.9296	0.9271	0.9264
0.98	0.9391	0.9162	0.9054	0.9368	0.9279	0.9271	0.9257
0.97	0.9370	0.9166	0.9035	0.9364	0.9272	0.9265	0.9242
0.96	0.9345	0.9147	0.9005	0.9360	0.9265	0.9265	0.9232
0.95	0.9304	0.9148	0.9010	0.9356	0.9243	0.9253	0.9217
0.94	0.9280	0.9133	0.8995	0.9335	0.9238	0.9255	0.9225
0.93	0.9259	0.9128	0.8986	0.9355	0.9237	0.9263	0.9214
0.92	0.9232	0.9120	0.8955	0.9327	0.9203	0.9237	0.9204
0.91	0.9183	0.9105	0.8947	0.9335	0.9208	0.9229	0.9194
0.90	0.9169	0.9090	0.8934	0.9303	0.9189	0.9226	0.9169
0.85	0.9013	0.9042	0.8840	0.9272	0.9109	0.9190	0.9127
0.80	0.8850	0.8973	0.8733	0.9222	0.9022	0.9138	0.9040
0.70	0.8462	0.8813	0.8477	0.9068	0.8780	0.9000	0.8851
0.60	0.7980	0.8565	0.8132	0.8866	0.8478	0.8806	0.8571
0.40	0.6762	0.7876	0.7195	0.8244	0.7585	0.8192	0.7735
0.20	0.5268	0.6827	0.5806	0.7284	0.6298	0.7237	0.6517
0.10	0.4649	0.6206	0.5112	0.6698	0.5639	0.6669	0.5889

† External moderator is modeled at 100%.

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

TABLE 6.4.6

MAXIMUM k_{eff} VALUES IN THE MPC-37 WITH UNDAMAGED (15x15F) AND DAMAGED FUEL

Bare Rod Array inside the DFC	Maximum k_{eff}, 4.0 wt%	Maximum k_{eff}, 5.0 wt%
8x8	0.8883	0.9122
10x10	0.8899	0.9135
12x12	0.8910	0.9152
14x14	0.8945	0.9177
15x15	0.8966	0.9198
16x16	0.8982	0.9224
17x17	0.9003	0.9238
18x18	0.9027	0.9262
20x20	0.9032	0.9276
22x22	0.9023	0.926
24x24	0.9008	0.9239

TABLE 6.4.7

MAXIMUM k_{eff} VALUES IN THE MPC-89 WITH UNDAMAGED (10x10A) AND DAMAGED FUEL

Bare Rod Array inside the DFC	Maximum k_{eff}, 4.8 wt% (planar average)
4x4	0.9389
6x6	0.9411
8x8	0.9432
9x9	0.9464
10x10	0.9454
11x11	0.9451
12x12	0.9460
13x13	0.9453
14x14	0.9444
16x16	0.9429
18x18	0.9423

TABLE 6.4.8

Not Used

TABLE 6.4.9

COMPARISON OF MCNP CONVERGENCE PARAMETERS

Calculation Parameters		**Maximum k_{eff}**	
Particles per Cycle	Skipped Cycles	**MPC-37 (17x17B, 5.0% ENRICHMENT)**	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
20,000	20	0.9380	0.9435
50,000	20	0.9376	0.9428
20,000	100	0.9387	0.9436
50,000	100	0.9379	0.9434

TABLE 6.4.10

COMPARISON OF MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 WITH CONDITIONS OF FILLED AND VOIDED GUIDE AND INSTRUMENT TUBES AT 5 % ENRICHMENT

Fuel Assembly Class	Maximum k_{eff}, Filled Tubes	Maximum k_{eff}, Voided Tubes
14x14A	0.8983	0.8887
14x14B	0.9172	0.9015
14x14C	0.9275	0.9277
15x15B	0.9311	0.9251
15x15C	0.9188	0.9134
15x15D	0.9421	0.9379
15x15E	0.9410	0.9365
15x15F	0.9455	0.9404
15x15H	0.9325	0.9317
15x15I	0.9357	0.9362
16x16A	0.9366	0.9320
17x17A	0.9194	0.9135
17x17B	0.9380	0.9292
17x17C	0.9424	0.9345
17x17D	0.9384	0.9293
17x17E	0.9392	0.9314

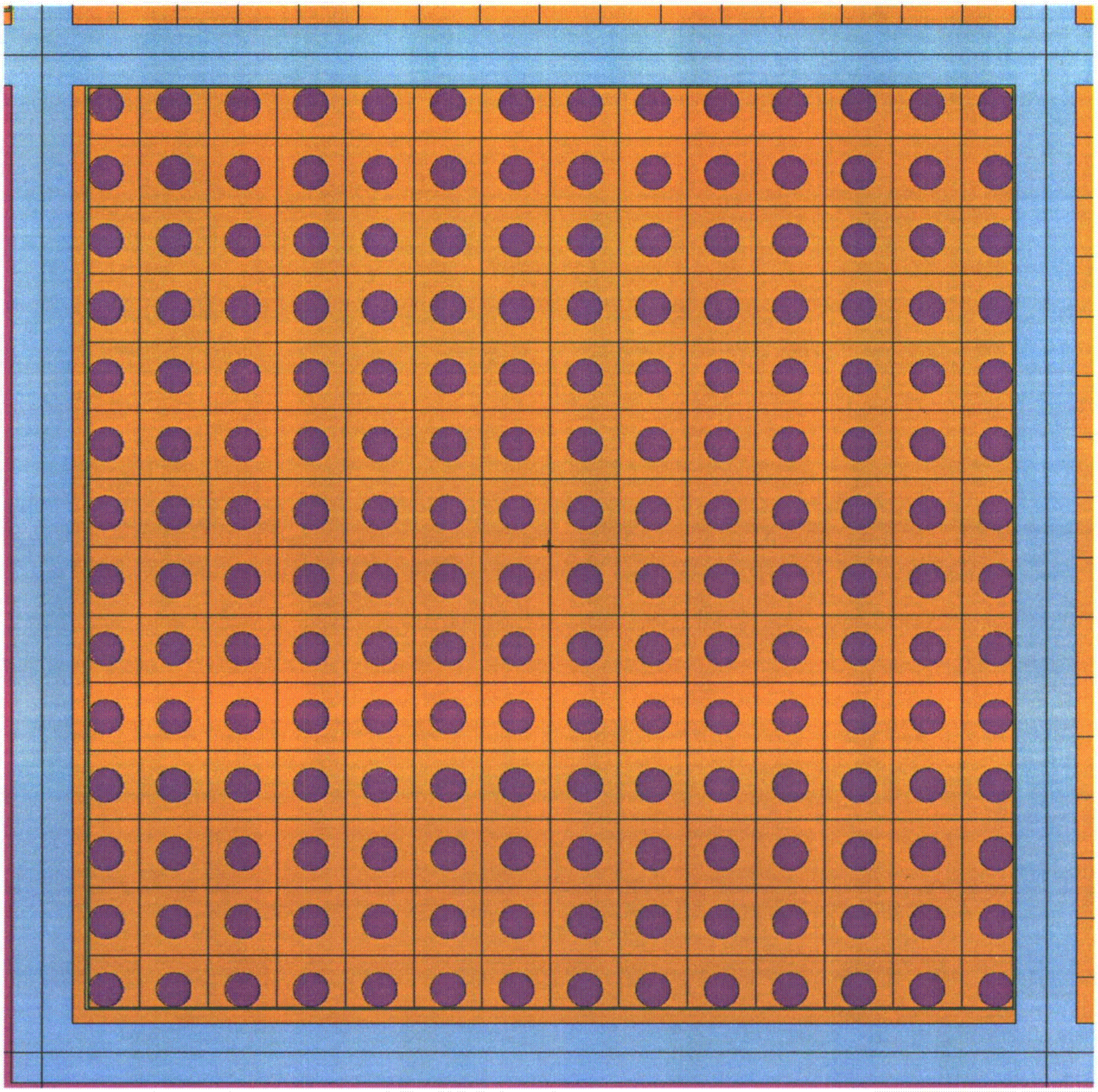

Figure 6.4.1: Calculational Model (planar cross-section) of a DFC in a MPC-37 cell with a 14x14 array of bare fuel rods

6.5 CRITICALITY BENCHMARK EXPERIMENTS

Benchmark calculations have been made on selected critical experiments, chosen, insofar as possible, to bound the range of variables in the cask designs. The most important parameters are (1) the enrichment, (2) cell spacing, and (3) the ^{10}B loading of the neutron absorber panels. Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included. No significant trends were evident in the benchmark calculations or the derived bias. Detailed benchmark calculations are presented in Appendix 6.A.

The benchmark calculations were performed with the same computer codes and cross-section data, described in Section 6.4, that were used to calculate the k_{eff} values for the cask. Further, all calculations were performed on the same computer hardware (personal computers).

6.6 REGULATORY COMPLIANCE

This section documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STORM FW system is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety, as well as the limiting fuel characteristics, are described in sufficient detail in this section to enable an evaluation of their effectiveness.

The HI-STORM FW system is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poison has shown that they will remain effective for a storage period greater than 60 years, and there is no credible way to lose it; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10CFR72.124(b).

The criticality evaluation has demonstrated that the cask will enable the storage of spent fuel for a minimum of 60 years with an adequate margin of safety. Further, the evaluation has demonstrated that the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, the HI-STORM FW system is in full compliance with the double contingency requirements of 10CRF72.124. Therefore, it is concluded that the criticality design features for the HI-STORM FW system are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The criticality evaluation provides reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel.

6.7 REFERENCES

[6.0.1] HI-STORM 100 FSAR, NRC Docket 72-1014, Holtec Report HI-2002444, Latest revision

[6.0.2] "Criticality Analyses for the HI-STORM FW System", Holtec Report HI-2094432 Rev.0 (proprietary)

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

[6.1.2] 10CFR72.124, "Criteria For Nuclear Criticality Safety."

[6.1.3] not used

[6.1.4] "MCNP - A General Monte Carlo N-Particle Transport Code, Version 5"; Los Alamos National Laboratory, LA-UR-03-1987 (2003).

[6.1.5] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

[6.1.6] "CASMO-4 Methodology", Studsvik/SOA-95/2, Rev. 0, 1995.

"CASMO-4 A Fuel Assembly Burnup Program, Users Manual," SSP-01/400, Rev. 1, Studsvik Scandpower, Inc., 2001.

"CASMO-4 Benchmark Against Critical Experiments", Studsvik/SOA-94/13, Studsvik of America, 1995.

[6.4.1] J.M. Cano, R. Caro, and J.M Martinez-Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," Nucl. Technol., 48, 251-260, (1980).

APPENDIX 6.A: BENCHMARK CALCULATIONS

Withheld in Accordance with 10 CFR 2.390

APPENDIX 6.B: MISCELLANEOUS INFORMATION

6.B.1 Sample Input File MPC-37

Withheld in Accordance with 10 CFR 2.390

6.B.2 Sample Input File MPC-89

Withheld in Accordance with 10 CFR 2.390

6.B.3 Analyzed Distributed Enrichment Patterns

Withheld in Accordance with 10 CFR 2.390

6.B.4 Assembly Cross Sections

Withheld in Accordance with 10 CFR 2.390

CHAPTER 7*: CONFINEMENT

7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STORM FW system is provided by the MPC. The design of the HI-STORM FW MPC assures that there are no credible design basis events that would result in a radiological release to the environment. The HI-STORM FW overpack and HI-TRAC VW transfer cask are designed to provide physical protection to the MPC during normal, off-normal, and postulated accident conditions to assure that the integrity of the MPC is maintained. The dry inert atmosphere in the MPC and the passive heat removal capabilities of the HI-STORM FW also assure that the SNF assemblies remain protected from long-term degradation.

A detailed description of the confinement structures, systems, and components important to safety is provided in Chapter 2. The structural adequacy of the MPC is demonstrated by the analyses documented in Chapter 3. The physical protection of the MPC provided by the overpack and the HI-TRAC Transfer Cask is demonstrated by the structural analyses documented in Chapter 3 for off-normal and postulated accident conditions that are considered in Chapter 11. The heat removal capabilities of the HI-STORM FW system are demonstrated by the thermal analyses documented in Chapter 4. Materials evaluation in Chapter 8 demonstrates the compatibility and durability of the MPC materials for long term spent fuel storage.

This chapter describes the HI-STORM FW confinement design and describes how the design satisfies the confinement requirements of 10CFR72 [7.0.1]. It also provides an evaluation of the MPC confinement boundary as it relates to the criteria contained in Interim Staff Guidance (ISG)-18 [7.0.2] and applicable portions of ANSI N14.5-1997 [7.0.3] as justification for reaching the determination that leakage from the confinement boundary is not credible and, therefore, a quantification of the consequence of leakage from the MPC is not required. This chapter is in general compliance with NUREG-1536 [7.0.4] as noted in Chapter 1.

It should be observed that the configuration of the confinement boundary of the MPCs covered by this FSAR is identical to that used in the MPCs in Docket No. 72-1014 (HI-STORM 100 system), including weld joint details and weld types and weld sizes. Therefore, it is reasonable to conclude that the safety evaluation conducted to establish confinement integrity in Docket No. 72-1014 is also applicable herein.

*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.

7.1 CONFINEMENT DESIGN CHARACTERISTICS

The confinement against the release of radioactive contents is the all welded MPC. There are no bolted closures or mechanical seals in the MPC confinement boundary.

The confinement boundary of the MPC consists of the following parts:

- MPC shell
- MPC base plate
- MPC lid
- MPC vent and drain port covers
- MPC closure ring
- associated welds

The combination of the welded MPC lid and the welded closure ring form the redundant closure of the MPC and satisfies the requirements of 10 CFR 72.236(e) [7.0.1]. The confinement boundary is shown in the licensing drawing package in Section 1.5. Chapter 2 provides design criteria for the confinement boundary. All components of the confinement boundary are important-to-safety, as specified on the licensing drawings. The MPC confinement boundary is designed, fabricated, inspected and tested in accordance with the applicable requirements of ASME Code, Section III, Subsection NB [7.1.1], with alternatives given in Chapter 2.

7.1.1 Confinement Vessel

The HI-STORM FW system confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The three major components of the MPC vessel are the shell, baseplate, and lid. The shell welds and the shell to baseplate weld are performed at the fabrication facility. The remaining confinement boundary welds are performed in the field (Table 7.1.1).

The MPC lid consists of two sections (referred to as upper and lower) welded together. Only the upper portion of the lid is credited in the confinement boundary. The lid is made intentionally thick by the addition of the lower portion of the lid to minimize radiation exposure to workers during MPC closure operations. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter.

Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by progressive liquid penetrant examinations (a multi-layer liquid penetrant examination), and a pressure test is performed. The MPC lid-to shell weld is not helium leakage tested since the weld meets the guidance of NRC Interim Staff Guidance (ISG)-15 [7.1.2] and criteria of ISG-18 [7.0.2], therefore leakage from the MPC lid-to-shell weld is not considered credible. Table 7.1.2 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one.

After the MPC lid weld is ensured to be acceptable the vent and drain port cover plates are welded in place, examined by the liquid penetrant method and a helium leakage test of each of the vent and drain port cover plate welds is performed. These welds are tested in accordance to the leakage test methods and procedures of ANSI N 14.5 [7.0.3] to the "leaktight" criterion of the standard. Finally, the MPC closure ring which also covers the vent and drain cover plates is installed, welded, and inspected by the liquid penetrant method. Chapters 9, 10, and 13 provide procedural guidance, acceptance criteria, and operating controls, respectively, for performance and acceptance of non-destructive examination of all welds made in the field.

After moisture removal and prior to sealing the MPC vent and drain ports, the MPC cavity is backfilled with helium. The helium backfill provides an inert, non-reactive atmosphere within the MPC cavity that precludes oxidation and hydride attack on the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and helps reduce the fuel cladding temperatures. The inert atmosphere in the MPC, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

The confinement boundary welds completed at the fabrication facility (i.e., the MPC longitudinal and circumferential shell welds and the MPC shell to baseplate weld) are referred to as the shop welds. After visual and liquid penetrant examinations, the shop welds are volumetrically inspected by radiography. The MPC shop welds are multiple-pass (6 to 8 passes) austenitic stainless steel welds. Helium leakage testing of the shop welds is performed as described in Table 10.1.1.

7.1.2 Confinement Penetrations

Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and also for MPC re-flooding during unloading operations. No other confinement penetrations exist in the MPC.

The MPC vent and drain ports are sealed by cover plates that are integrally welded to the MPC lid. No credit is taken for the sealing action provided by the vent and drain port cap joints. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, providing the redundant closure of these penetrations. The redundant closure of the MPC satisfies the requirements of 10CFR72.236(e) [7.0.1].

7.1.3 Seals and Welds

Section 7.1.1 describes the design of the confinement boundary welds. The welds forming the confinement boundary is summarized in Table 7.1.1.

The use of multi-pass welds with surface liquid penetrant inspection of root, intermediate, and final passes renders the potential of a leak path through the weld between the MPC lid and the shell to be non-credible. The vent and drain port cover plate welds are helium leak tested in the field, providing added assurance of weld integrity. Additionally after fuel loading, a Code pressure test is performed on the MPC lid-to-shell weld to confirm the structural integrity of the weld.

The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or other failure modes such as hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld will not result in release of radioactive material to the environment. Section 10.1 provides a summary of the closure weld examinations and tests.

7.1.4 Closure

The MPC is an integrally welded pressure vessel without any unique or special closure devices. All closure welds are examined using the liquid penetrant technique to ensure their integrity. Additionally, the vent and drain port cover plate welds are each helium leakage tested to be "leaktight" in accordance with the leakage test methods and procedures of ANSI N14.5-1997 [7.0.3]. Since the MPC uses an entirely welded redundant closure system with no credible leakage, no direct monitoring of the closure is required.

Table 7.1.1

MPC CONFINEMENT BOUNDARY WELDS

MPC Weld Location	Weld Type†	ASME Code Category (Section III, Subsection NB)
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port covers to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C
MPC closure ring to closure ring radial weld	Partial Penetration Groove (field weld)	††

† The tests and inspections for the Confinement Boundary welds are listed in Section 10.1

†† This joint is governed by NB-5271 (liquid penetrant examination).

Table 7.1.2

COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE

DESIGN/QUALIFICATION GUIDANCE	HOLTEC MPC DESIGN
The canister is constructed from austenitic stainless steel.	The MPC enclosure vessel is constructed entirely from austenitic stainless steel (Alloy X). Alloy X is defined as Type 304, 304LN, 316, or 316LN material.
The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3.	The MPC lid-to-shell closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8" of weld depth, which corresponds to the critical flaw size.
The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents and natural phenomena as required in 10CFR72.	The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is designed to ensure that the Confinement Boundary will not leak during any credible accident event and under the non-mechanistic tip-over scenario.
Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10CFR72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.	Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STORM FW CoC. Compliance with 10CFR72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.
Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.	The NRC has approved Holtec's Quality Assurance program under 10CFR71. That same QA program has been adopted for activities governed by 10CFR72 as permitted by 10 CFR 72.140(d)

7.2 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

Once sealed and transferred into the HI-STORM FW overpack there is no mechanism under normal and off-normal conditions of storage for the confinement boundary to be breached. Chapter 3 shows that all confinement boundary components are maintained within their Code-allowable stress limits during normal and off-normal storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal and off-normal conditions of storage. Since the MPC confinement vessel remains intact, the design temperatures and pressure are not exceeded, and leakage from the MPC confinement boundary as discussed in Section 7.1 is not credible, there can be no release of radioactive material during normal and off-normal conditions of storage.

The MPC is dried and helium backfilled prior to sealing and no significant moisture or other gases remain inside the MPC. Therefore, a credible mechanism for any radiolytic decomposition that could cause an increase in the MPC internal pressure is absent. The potential for the explosive level of gases due to radiological decomposition in the MPC is eliminated by excluding foreign materials in the MPC.

7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The analysis in Chapter 3 and results discussed in Chapter 12 demonstrates that the MPC remains intact during and after all postulated accident conditions; therefore there can be no release of radioactive material causing any additional dose contribution to the site boundary during these events.

7.4 REFERENCES

[7.0.1] 10CFR72, Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater than Class C Waste," USNRC, Washington, DC.

[7.0.2] Interim Staff Guidance-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," USNRC, Washington, DC, May 2003.

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

[7.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, Washington, DC, January, 1997.

[7.1.1] ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1 Components, American Society of Mechanical Engineers, New York, NY, 2007 Edition.

[7.1.2] Interim Staff Guidance-15, "Materials Evaluation", USNRC, Washington, DC, January 2001.

[7.1.3] Holtec Proprietary Report HI-2022850, Revision 0, "Summary Report on MPC Leak Tightness Test", April 2002.

CHAPTER 8: MATERIAL EVALUATION

8.1 INTRODUCTION

This chapter presents an assessment of the materials selected for use in the HI-STORM FW System components identified in the licensing drawings in Section 1.5. In this chapter and Chapter 3 of this FSAR, the significant mechanical, thermal, radiological and metallurgical properties of materials identified for use in the components of the HI-STORM FW System are presented. This chapter focuses on the HI-STORM FW material properties to assess compliance with the ISG-15 [8.1.1] and ISG-11 [8.1.2] requirements. The principal purpose of ISG-15 is to evaluate the dry cask storage system to ensure adequate material performance of the independent spent fuel storage installation (ISFSI) components designated as important to safety under normal, off-normal and accident conditions. Some areas of review applicable to the suitability assessment of the materials have been addressed elsewhere in this FSAR and are referenced from this chapter as necessary. Areas that require further details are reviewed within this chapter as necessary to satisfy the requirements of ISG-15. Guidance on performing the review is adopted directly from ISG-15 and ISG-11.

ISG-15 sets down the following general acceptance criteria for material evaluation.

- The safety analysis report should describe all materials used for dry spent fuel storage components designated as important to safety, and should consider the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness in relation to all safety functions.

- The dry spent fuel storage system should employ materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials should not degrade to the extent that a safety concern is created.

The information compiled in this chapter addresses the above acceptance criteria. To perform the material suitability evaluation, it is necessary to characterize the following for each component: (i) the applicable environment, (ii) the potential degradation modes and (iii) the potential hazards to continued effectiveness of the selected material.

The operating environments of the different components of the cask system are not the same. Likewise, the potential degradation modes and hazards are different for each component. Tables 8.1.1, 8.1.2, and 8.1.3 provide a summary of the environmental states, potential degradation modes and hazards applicable to the MPC, the HI-STORM FW overpack and the HI-TRAC VW cask, respectively. The above referenced tables serve to guide the material suitability evaluation for the HI-STORM FW System.

To provide a proper context for the subsequent evaluations, the potential degradation mechanisms applicable to the ventilated systems are summarized in Table 8.1.4. The degradation mechanisms listed in Table 8.1.4 are considered in the suitability evaluation presented later in this chapter.

The material evaluation presented in this chapter is intended to be complete, even though a conclusion of the adequacy of the materials can be made on the strength of the following facts:

i. The materials used in HI-STORM FW are, with the sole exception of Metamic-HT, identical to those used in the widely deployed HI-STORM 100 System (Docket No. 72-1014).

ii. The thermal environment in the HI-STORM FW system emulates the HI-STORM 100 system in all respects.

iii. The fuel loading and short-term operations are essentially identical to those that have been practiced in the HI-STORM 100 system throughout the industry.

Table 8.1.1

CONSIDERATIONS GERMANE TO THE MPC MATERIAL PERFORMANCE

Consideration	Short-Term Operations	Long-Term Storage
Environment	Aqueous (with Boric acid in PWR plants), characterized by moderately hot (<212°F) water during fuel loading, rapid evaporation during welding and drying operations	MPC's internal environment is hot (≤752°F), inertized and dry. Temperature of the MPC cycles very gradually due to changes in the environmental temperature.
Potential degradation modes	Hydrogen generation from oxidation of aluminum and aluminum alloy internals. Risk to the integrity of fuel cladding from thermal transients caused by vacuum drying.	Corrosion of the external surfaces of the MPC (stress, corrosion, cracking, pitting, etc.)
Potential hazards to effective performance	Inadequate drying of waterlogged fuel rods.	Blockage of ventilation ducts under an extreme environmental phenomenon leading to a rapid heat-up of the MPC internals.

Table 8.1.2

CONSIDERATIONS GERMANE TO THE HI-STORM FW OVERPACK MATERIAL PERFORMANCE*

Consideration	Performance Data
Environment	Cool ambient air progressively heated as it rises in the overpack/MPC annulus heating the inside surface of the cask. The heated air has reduced relative humidity. Direct heating of the overpack inner shell by radiation prevented by the "heat shields" is described in Chapter 1. External surface of the overpack including the top lid is heated and in contact with ambient air, rain, and snow, as applicable.
Potential degradation modes	Peeling or perforation of surface preservatives and corrosion of any exposed steel surfaces.
Potential hazards to effective performance	Blockage of ducts by debris leading to overheating of the concrete in the overpack, scorching of the cask by proximate fire, lightning.

* Short-term operations are not applicable to the HI-STORM FW overpack.

Table 8.1.3

CONSIDERATIONS GERMANE TO THE OF HI-TRAC VW MATERIAL PERFORMANCE*

Consideration	Performance Data
Environment	Heated fuel pool water on the outside and demineralized water in contact with the inside surface, heated water in the "water jacket". Temperature ramps on the inside surface during the drying and "backfill" operation.
Potential degradation modes	Peeling or perforation of surface coatings, loss of effectiveness of bottom lid gasket.
Potential hazards to effective performance	Lead slump due to sudden inertial loading; contamination of the inside surface of the cask by pool water, partial loss of heat rejection resulting in boiling of water in the water jacket, impact from tornado missile during transfer to the ISFSI.

* Long-term storage conditions are not applicable to the transfer cask.

Table 8.1.4

FAILURE AND DEGRADATION MECHANISMS APPLICABLE TO VENTILATED SYSTEMS[§]

	Mechanism	Area of Performance Affected	Vulnerable Parts
1.	General Corrosion	Structural capacity	All carbon steel parts
2.	Hydrogen Generation	Personnel safety during short-term operations	Coatings, parts made of aluminum or aluminum alloys
3.	Stress Corrosion Cracking	Structural	Austenitic Stainless Steel
4.	Creep	Criticality control	Fuel Basket
5.	Galling	Equipment handling and deployment	Threaded Fasteners
6.	Hysteresis	During fuel loading in the pool	HI-TRAC VW Bottom Lid Gaskets
7.	Fatigue	Structural Integrity	Fuel Cladding & Bolting
8.	Brittle Fracture	Structural Capacity	Thick Steel Parts
9.	Boron Depletion	Criticality Control	Neutron Absorber

[§] This table lists all potential (generic) mechanisms, whether they are credible for the HI-STORM FW System or not. The viability of each failure mechanism is discussed later in this chapter.

8.2 MATERIAL SELECTION

The following acceptance criteria are applicable for material selection per ISG-15.

a. The material properties of a dry spent fuel storage component should meet its service requirements in the proposed cask system for the duration of the licensing period.

b. The materials that comprise the dry spent fuel storage should maintain their physical and mechanical properties during all conditions of operations. The spent fuel should be readily retrievable without posing operational safety problems.

c. Over the range of temperatures expected prior to and during the storage period, any ductile-to-brittle transition of the dry spent fuel storage materials, used for structural and nonstructural components, should be evaluated for its effects on safety.

d. Dry spent fuel storage gamma shielding materials (e.g. lead) should not experience slumping or loss of shielding effectiveness to an extent that compromises safety. The shield should perform its intended function throughout the licensed service period.

e. Dry spent fuel storage materials used for neutron absorption should be designed to perform their safety function.

f. Dry spent fuel storage protective coatings should remain intact and adherent during all loading and unloading operations within wet or dry spent fuel facilities, and during long-term storage.

The above criteria have been utilized in selecting the material types for the HI-STORM FW system. The selected materials provide the required heat transfer, confinement, shielding and the criticality control of the stored spent fuel and are capable of withstanding loadings including seismic, temperature cycles due to internal heat and ambient temperature variation, extreme temperature conditions, loads due to natural phenomena like tornado missiles, flooding and other credible hypothetical accident scenarios. The HI-STORM FW components must withstand the environmental conditions experienced during normal operation, off-normal conditions and accident conditions for the entire service life.

The selection of materials is guided by the applicable loadings and potential failure modes. An emphasis has been placed on utilizing proven materials that have established properties and characteristics and are of proven reliability. Where a relatively new material (e.g., Metamic-HT) is used, comprehensive tests have been conducted to ensure reliability.

The major structural materials used in HI-STORM FW System are discussed in this section. The mechanical and thermal properties of these materials are presented in Section 8.4. The materials for welds are discussed in Section 8.5. The structural materials for bolts and fasteners are discussed in Section 8.6. Coatings and paints are discussed in Section 8.7. Gamma and neutron shielding materials are treated in Section 8.8. The neutron absorbing materials are discussed in Section 8.9.

Chapter 1 provides a general description of the HI-STORM FW System including information on materials of construction. All materials of construction are identified in the drawing package provided in Section 1.5 and the ITS categories of the sub-components are identified in Table 2.0.1 through 2.0.8.

8.2.1 Structural Materials

8.2.1.1 Cask Components and Their Constituent Materials

The major structural materials that are used in the HI-STORM FW System are Alloy X, Metamic-HT, carbon steel, and aluminum. They are further discussed below in light of the ISG-15 requirements.

MPC

All structural components in an MPC Enclosure Vessel are made of Alloy X (stainless steel). Appendix 1.A provides discussions on Alloy X materials. The fuel basket is made of Metamic-HT neutron absorber described in Appendix 1.B. The confinement boundary is made of stainless steel material for its superior strength, ductility, and resistance to corrosion and brittle fracture for long term storage. The basket shims used to support the basket are made of a creep resistant aluminum alloy. The two-piece MPC lid is either made entirely of Alloy X or the bottom portion of the lid is made of carbon steel with stainless steel veneer. The principal materials used in the fabrication of the MPC are listed in Section 1.2.

HI-STORM

The main structural function of the overpack is provided by carbon steel and the main shielding function is provided by plain concrete. Chapter 1 presents discussions on these materials. The materials used in the fabrication of the overpack are listed in Section 1.2.

HI-TRAC

As discussed in Chapter 1, the HI-TRAC VW transfer cask is principally made of carbon steel and lead. The HI-TRAC VW is equipped with a water jacket. The materials used in the fabrication of the transfer cask are listed in Section 1.2.

8.2.1.2 Synopsis of Structural Materials

i. Alloy X

The MPC enclosure vessel design allows use of any one of the four Alloy X materials: Types 304, 304LN, 316 and 316LN. Qualification of structures made of Alloy X is accomplished by using the least favorable mechanical and thermal properties of the entire group for all MPC mechanical, structural, neutronic, radiological, and thermal conditions. Each of these material properties are provided in the ASME Code Section II [8.3.1].

As discussed in Appendix 1.A, the Alloy X approach is conservative because, no matter which material is ultimately utilized, the Alloy X guarantees that the performance of the MPC will meet or exceed the analytical predictions. The material properties are provided at various temperatures.

All structural analyses utilize conservatively established material properties such as design stress intensity, tensile strength, yield strength, and coefficient of thermal expansion for the range of temperature conditions that would be experienced by the cask components.

Chapter 3 provides the structural evaluation for the MPC Enclosure Vessel which is made of Alloy X. It is demonstrated that Alloy X provides adequate structural integrity for the MPC enclosure vessel under normal, off normal, and accident conditions. As shown in Chapter 4, the maximum metal temperature for Alloy X for the Confinement Boundary remains the design temperatures in Table 2.2.3 under all service modes. As shown in ASME Code Case N-47-33 (Class 1 Components in Elevated Temperature Service, 1995 Code Cases, Nuclear Components), the strength properties of austenitic stainless steels do not change due to exposure to 1000°F temperature for up to 10,000 hours.

Since stainless steel materials do not undergo a ductile-to-brittle transition in the minimum permissible service temperature range of the HI-STORM FW System, brittle fracture is not a concern for the MPC components. Subsection 8.4.3 presents further discussions on brittle fracture.

In Section 8.12, the potential for chemical and galvanic reaction of Alloy X in short-term and long-term operating conditions is evaluated. Alloy X is also used in the Confinement Boundary of all HI-STORM 100 MPCs.

ii. Metamic-HT

Criticality control in the HI-STORM FW System is provided by the coplanar grid work of the Fuel Basket honeycomb, made entirely of the Metamic-HT extruded metal matrix composite plates. The boron in Metamic-HT provides criticality control in the HI-STORM FW System. The Metamic-HT neutron absorber is a successor to the Metamic (classic) product widely used in dry storage fuel baskets and spent fuel storage racks (the "HT" designation in Metamic-HT stands for high temperature and is derived from this characteristic). Metamic-HT has been licensed in the HI-STAR 180 transport cask (Docket No. 71-9325).

Metamic-HT is also engineered to possess the necessary mechanical characteristics for structural application. The mechanical properties of Metamic-HT are derived from the strengthening of its aluminum matrix with ultra fine-grained (nano-particle size) alumina (Al_2O_3) particles that anchor the grain boundaries for high temperature strength and creep resistance.

Critical properties of Metamic-HT have been established as minimum guaranteed values by conducting tests [8.9.7] using ASTM sanctioned procedures (See Appendix 1.B.). The structural properties include yield strength, tensile strength, Young's modulus, elongation, Charpy impact energy, area reduction, and creep strain.

The neutron absorbing properties of Metamic-HT are addressed in Section 8.9 and also in Appendix 1.B.

Chapter 3 presents structural evaluation of spent fuel basket made of Metamic-HT wherein it is concluded that the Metamic-HT plates possess adequate structural strength to meet the loadings postulated for the fuel basket. Section 8.12 presents potential for chemical and galvanic reaction in Metamic-HT under short-term and long-term operating conditions.

All Metamic-HT material procured for use in the Holtec casks is qualified as *important-to-safety* (ITS). Accordingly, material and manufacturing control processes are established to eliminate the incidence of errors, and inspection steps are implemented to serve as an independent set of barriers to ensure that all *critical characteristics* defined for the material by the cask designer are met in the manufactured product. Additional discussions on the manufacturing of Metamic-HT are provided in Appendix 1.B and also in Chapter 10.

iii. Carbon Steel, Low-Alloy, and Nickel Alloy Steel

Materials for HI-STORM FW overpack and HI-TRAC VW transfer cask including the parts used to lift the overpack and the transfer cask, which may also be referred to as "significant-to-handling" or "STH" parts, are selected to preclude any concern of brittle fracture. Details of discussions are provided in Subsection 8.4.3.

Steel forging materials for low temperature applications have been selected for the STH components that have thicknesses greater than 2" so that acceptable fracture toughness at low temperatures can be assured. All other major steel structural materials in the HI-STORM FW overpack and HI-TRAC VW cask are made of fine grain low carbon steel (see drawings in Section 1.5).

The mechanical properties of these materials are provided in Section 3.3. Section 3.1 provides allowable stresses under different loading conditions and impact testing requirements for these materials.

Chapter 3 provides structural evaluations of the HI-STORM FW System components. It is demonstrated that the structural steel components of the HI-STORM FW overpack meet the allowable stress limits for normal, off-normal and accident loading conditions.

8.2.2 Nonstructural Materials

i. Aluminum Alloy

The space between the fuel basket and the inside surface of the Confinement Boundary is occupied by specially shaped precision machined basket shims made of a high strength and creep resistant aluminum alloy. The basket shims establish a conformal contact interface with the fuel basket and the MPC shell, and thus prevent significant movement of the basket. The basket shims are extruded and machined to a precise shape with a high degree of accuracy.

The clearance between the basket shims and the interfacing machined surface of the MPC cavity is set to be sufficiently small such that the thermal expansion of the parts inside the MPC under Design Basis heat load conditions will minimize any macro-gaps at the interface and thus minimize any resistance to the outward flow of heat, while ensuring that there is no restraint of free thermal expansion.

To further enhance thermal performance, the aluminum alloy basket shims are hard anodized. This provides for added corrosion protection and to achieve the emissivity value specified in Section 4.2. Mechanical properties of the shim material are provided in Section 3.3.

The basket shim material utilized in the HI-STORM FW system has also been used in other casks (viz. HI-STAR 180).

ii. Concrete

The plain concrete between the overpack inner and outer steel shells and in the overpack lid is specified to provide the necessary shielding properties and compressive strength. Appendix 1.D of the HI-STORM 100 FSAR which provides technical and placement requirements on plain concrete is also invoked for HI-STORM FW concrete.

The HI-STORM FW overpack concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates and does not require rebar. As the HI-STORM FW overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete.

The technical requirements on testing and qualification of the HI-STORM FW plain concrete are identical to those used in the HI-STORM 100 program. Accordingly, the testing and placement guidelines in Appendix 1.D of the HI-STORM 100 FSAR (Docket No. 72-1014), is incorporated in this SAR by reference.

ACI 318 is the reference code for the plain concrete in the HI-STORM FW overpack. ACI 318.1-85(05) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited in structural analysis.

The gamma shielding characteristics of concrete is considered in Section 8.8.

iii. Lead

HI-TRAC VW contains lead between its inner and the middle shell for gamma shielding. The load carrying capacity of lead is neglected in all structural analysis. However, in the analysis of a tornado missile strike the elasto-plastic properties of lead are considered in characterizing the penetration action of the missile.

Applicable mechanical properties of lead are provided in Section 3.3. Shielding properties of lead are provided in Section 8.8.

8.2.3 Critical Characteristics and Equivalent Materials

As defined in the Glossary, the *critical characteristics* of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function. However, material designations adopted by the International Standards Organization (ISO) also affect the type of steels and steel alloys available from suppliers around the world. Therefore, it is necessary to provide for the ability in this FSAR to substitute materials with equivalent materials in the manufacture of the equipment governed by this FSAR.

As defined in the Glossary, *equivalent materials* are those materials with critical characteristics that meet or exceed those specified for the designated material. Substitution by an equivalent material can be made after the equivalence in accordance with the provisions of this FSAR has been established.

The concept of equivalent materials explained above has been previously used in this FSAR to qualify four different austenitic stainless steel alloys (ASME SA240 Types 304, 304LN, 316, and 316LN) to serve as candidate MPC materials.

The equivalence of materials is directly tied to the notion of *critical characteristics*. A critical characteristic of a material is a material property whose value must be specified and controlled to ensure an SSC will render its intended function. The numerical value of the critical characteristic invariably enters in the safety evaluation of an SSC and therefore its range must be guaranteed. To ensure that the safety calculation is not adversely affected properties such as Yield Strength, Ultimate Strength and Elongation must be specified as *minimum* guaranteed values. However, there are certain properties where both minimum and maximum acceptable values are required (in this category lies specific gravity and thermal expansion coefficient).

Table 8.2.1 lists the array of properties typically required in safety evaluation of an SSC in dry storage and transport applications. The required value of each applicable property, guided by the safety evaluation needs defines the critical characteristics of the material. The subset of applicable properties for a material depends on the role played by the material. The role of a material in the SSC is divided into three categories:

Type	Technical Area of Applicability
S	Those needed to ensure structural compliance
T	Those needed to ensure thermal compliance (temperature limits)
R	Those needed to ensure radiation compliance (criticality and shielding)

The properties listed in Table 8.2.1 are the ones that may apply in a dry storage or transport application.

The following procedure shall be used to establish acceptable equivalent materials for a particular application.

Criterion i: Functional Adequacy:
Evaluate the guaranteed critical characteristics of the equivalent material against the values required to be used in safety evaluations. The required values of each critical characteristic must be met by the minimum (or maximum) guaranteed values (MGVs of the selected material).

Criterion ii: Chemical and Environmental Compliance:
Perform the necessary evaluations and analyses to ensure the candidate material will not excessively corrode or otherwise degrade in the operating environment.

A material from another designation regime that meets Criteria (i) and (ii) above is deemed to be an acceptable material, and hence, equivalent to the candidate material.

Equivalent materials as an alternative to the U.S. national standards materials (e.g., ASME, ASTM, or ANSI) shall not be used for the Confinement Boundary materials. Equivalent materials as alternative to Holtec's specialty engineered Metamic-HT material shall not be used for the MPC fuel basket. For other ITS materials, recourse to equivalent materials shall be made only in the extenuating circumstances where the designated material in this FSAR is not readily available.

As can be ascertained from its definition in the glossary, the *critical characteristics* of the material used in a subcomponent depend on its function. The overpack lid, for example, serves as a shielding device and as a physical barrier to protect the MPC against loadings under all service conditions, including extreme environmental phenomena. Therefore, the critical characteristics of steel used in the lid are its strength (yield and ultimate), ductility, and fracture resistance.

The appropriate critical characteristics for structural components of the HI-STORM FW System, therefore, are:

i. Material yield strength, σ_y
ii. Material ultimate strength, σ_u
iii. Elongation, ε
iv. Charpy impact strength at the lowest service temperature for the part, C_i

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than their minimum values used in the qualifying analyses used in this FSAR. The above *critical characteristics* apply to all materials used in the primary and secondary structural parts of the steel weldment in the overpack.

In the event that one or more of the *critical characteristics* of the replacement material is slightly lower than the original material, then the use of the §72.48 process shall be necessary to ensure that all regulatory predicates for the material substitution are fully satisfied.

Table 8.2.1
Critical Characteristics of Materials Required for Safety Evaluation of Storage and Transport Systems

	Property	Type	Purpose	Bounding Acceptable Value
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of $\delta_{min.}$, %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at ambient conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum allowable creep rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Thermal conductivity (minimum averaged value in the range of ambient to maximum service temperature, t_{max})	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
8.	Minimum Emissivity	T	To ensure that the thermal calculations are performed conservatively	Min.
9.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
10.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. (and Max.)
11.	Boron-10 Content	R	To control reactivity	Min.

8.3 APPLICABLE CODES AND STANDARDS

The principle codes and standards applied to the HI-STORM FW System components are the ASME Boiler and Pressure Vessel Code [8.3.1], the ACI code [8.3.2], the ASTM Standards and the ANSI standards. Chapter 1 provides details of the specific applications of these codes and standards along with the other codes and standards that are applicable.

Section 1.0 of this FSAR provides a tabulation of this FSAR's compliance with NUREG-1536. This section also provides a list of clarifications and alternatives to NUREG-1536. This list of clarifications and alternatives discusses Holtec International's approach for compliance with the underlying intent of the guidance and also provides the justification for the alternative method for compliance adopted in this FSAR. Section 1.2 identifies the ASME code paragraphs applicable for the design of the HI-STORM FW overpack primary load bearing parts, summarizes the code requirements for the fabrication of the HI-STORM FW components, and refers to the national standards (e.g., ASTM, AWS, ANSI, etc.) used for the material procurement and welding.

Chapter 2 discusses factored load combinations for ISFSI pad design per NUREG-1536 [8.3.3], which is consistent with ACI-349-85. Codes ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R, "Guide for Concrete Floor and Slab Construction"; and ACI 224R-90, "Control of Cracking in Concrete Structures" are also used in the design and construction of the concrete pad. Section 2.2 elaborates on the specific applications of ASME Boiler and Pressure Vessel code and provides a list of ASME code alternatives for the HI-STORM FW System.

Section 3.1 provides allowable stresses and stress intensities for various materials extracted from applicable ASME code sections for various service conditions. This section also provides discussions on fracture toughness test requirements per ASME code sections. Mechanical properties of materials are extracted from applicable ASME sections and are tabulated for various materials used in HI-STORM FW System. Concrete properties are from ACI 318-89 code. Section 3.7 presents discussions on compliance on NUREG-1536 and stipulations of 10CFR72 requirements to provide reasonable assurance with respect to the adequacy of the HI-STORM FW System.

In order to meet the requirements of the codes and standards the materials must conform to the minimum acceptable physical strengths and chemical compositions and the fabrication procedures must satisfy the prescribed requirements of the applicable codes.

Additional codes and standards applicable to welding are discussed in Section 8.5 and those for the bolts and fasteners are discussed in Section 8.6.

Review of the above shows that the identified codes and standards are appropriate for the material control of major components. Additional material control is identified in material specifications. Material selections are appropriate for environmental conditions to be encountered during loading, unloading, transfer and storage operations. The materials and fabrication of major components are suitable based on the applicable codes of record.

8.4 MATERIAL PROPERTIES

This section provides discussions on material properties that mainly include mechanical and thermal properties. The material properties used in the design and analysis of the HI-STORM FW System are obtained from established industry codes such as ASME Boiler and Pressure Vessel Code [8.4.1], ASTM publications, handbooks, textbooks, other NRC-reviewed SARs, and government publications, as appropriate.

8.4.1 Mechanical Properties

Section 3.3 presents mechanical properties of materials used in the HI-STORM FW System. The structural materials include Alloy X, Metamic-HT, carbon steel, low-alloy and nickel-alloy steel, bolting materials and weld materials. The properties include yield stress, mean coefficient of thermal expansion, ultimate stress and the Young's modulus of these materials and their variations with temperature. Certain mechanical properties are also provided for nonstructural materials such as concrete and lead used for shielding. Additional properties of the neutron absorbing material Metamic-HT are discussed in Section 8.9.

The discussion on mechanical properties of materials in Chapter 3 provides reasonable assurance that the class and grade of the structural materials are acceptable under the applicable construction code of record. Selected parameters such as the temperature dependent values of stress allowables, modulus of elasticity, Poisson's ratio, density, thermal conductivity and thermal expansion have been appropriately defined in conjunction with other disciplines. The material properties of all code materials are guaranteed by procuring materials from Holtec approved vendors through material dedication*, process if necessary.

8.4.2 Thermal Properties

Section 4.2 presents thermal properties of materials used in the MPC such as Alloy X, Metamic-HT, aluminum shims and helium gas; materials present in HI-STORM FW such as carbon steel and concrete; and materials present in HI-TRAC VW transfer cask that include carbon steel, lead and demineralized water. The properties include density, thermal conductivity, heat capacity, viscosity, and surface emissivity/absorptivity. Variations of these properties with temperature are also provided in tabular forms.

The thermal properties of fuel (UO_2) and fuel cladding are also reported in Section 4.2.

Thermal properties are often obtained from standard handbooks and established text books (see Table 4.2.1). When variations of thermal properties are observed the most conservative values are established as input for the design of the components of the HI-STORM FW System.

* A term of art in nuclear quality assurance.

8.4.3 Low Temperature Ductility of Ferritic Steels*

The risk of brittle fracture in the HI-STORM FW components is eliminated by utilizing materials that maintain high fracture toughness under extremely cold conditions.

The MPC canister is constructed from a menu of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum service temperature range of the HI-STORM FW System. Therefore, brittle fracture is not a concern for the MPC components.

Such an assertion cannot be made *a' priori* for the HI-STORM FW storage overpack and HI-TRAC VW transfer cask that contain ferritic steel parts. In general, the impact testing requirement for the HI-STORM FW overpack and the HI-TRAC VW transfer cask is a function of two parameters: the Lowest Service Temperature (LST)† and the normal stress level. The significance of these two parameters, as they relate to impact testing of the overpack and the transfer cask, is discussed below.

In normal storage mode, the LST of the HI-STORM FW storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). However, during the HI-STORM FW overpack transport operations, the applicable lowest service temperature is per 0°F (per the Technical Specifications). Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM FW System; namely,

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

LST = -40°F for the HI-STORM FW overpack during storage operations.

SA350-LF2 and SA350-LF3 have been selected as the material for the STH parts due to their capability to maintain acceptable fracture toughness at low temperatures (see Table 5 in SA350 of ASME Section IIA).

Table 3.1.9 provides a summary of impact testing requirements for the materials used in the HI-STORM FW System to ensure prevention of brittle fracture.

8.4.4 Creep Properties of Materials

Creep, a visco-elastic and visco-plastic effect in metals, manifests itself as a monotonically increasing deformation if the metal part is subjected to stress under elevated temperature. Since

* This subsection has been copied from the HI-STORM 100 FSAR (Section 3.1) without any substantive change.
† LST (Lowest Service Temperature) is defined as the daily average for the host ISFSI site when the outdoors portions of the "short-term operations" are carried out.

certain parts of the HI-STORM FW System, notably the fuel basket, operate at relatively high temperatures, creep resistance of the fuel basket is an important property. Creep is not a concern in the MPC enclosure vessel, the HI-STORM FW overpack, or the HI-TRAC VW steel weldment because of the operating metal temperatures, stress levels and material properties. Steels used in ASME Code pressure vessels have a high threshold temperature at which creep becomes a factor in the equipment design. The ASME Code Section II material properties provide the acceptable upper temperature limit for metals and alloys acceptable for pressure vessel service. In the selection of steels for the HI-STORM FW System, a critical criterion is to ensure that the sustained metal temperature of the part made of the particular steel type shall be less than the Code allowable temperature for pressure vessel service (ASME Section III Subsection ND). This criterion guarantees that excessive creep deformation will not occur in the steels used in the HI-STORM FW System.

As discussed below, the incidence of creep in the fuel basket is a not a trivial matter because lateral creep deformation can alter the reactivity control characteristics of the basket.

8.4.4.1 Metamic-HT

Metamic-HT is the sole constituent material in the HI-STORM FW fuel basket. The suitability of Metamic-HT for the conditions listed in Table 8.1.1 are considered in the "Metamic-HT Qualification Sourcebook" [8.9.7], submitted in USNRC Docket No. 71-9325 as a Holtec proprietary document.

The Metamic Sourcebook contains data on the testing to determine the creep characteristics of the Metamic-HT under both unirradiated and irradiated conditions. A creep equation to estimate a bounding estimate of total creep as a function of stress and temperature is also provided. The creep equation developed from this test provides a conservative prediction of accumulated creep strain by direct comparison to measured creep in unirradiated and irradiated coupons.

The creep equation for Metamic-HT that bounds *all* measured data (tests run for 20,000 hours) is of the classical exponential form in stress and temperature (see Appendix 1.B) stated symbolically $\varepsilon = f(\sigma,T)$.

Creep in the fuel basket will not affect reactivity because the basket is oriented vertically during all operations. The lateral loading of the fuel basket walls is insignificant and hence there is no mechanistic means for the basket panels to undergo lateral deformation from creep.

The creep effect would tend to shorten the fuel basket under the self-weight of the basket. An illustrative calculation of the cumulative reduction of the basket length is presented below to demonstrate the insignificant role of creep in the fuel basket.

The in-plane compressive stress, σ, at height x in the basket panel is given by

$$\sigma = \rho(H-x) \tag{8.1}$$

Where
ρ = density of Metamic-HT
H = height of the fuel basket

Using the above stress equation, the total creep shrinkage, δ, is given by

$$\delta = \int_o^{\tau^*} \{\int_o^H (\sigma, T)\, dx\} d\tau \tag{8.2}$$

Where
T = panel's metal temperature, initial value conservatively assumed to be 350°C (from Section 4.6) and dropping linearly to 150°C at 60 years.
τ^* = 60 years
H = height of the basket (approximately 200 inches)

Using the creep equation (provided in Appendix 1.B) and performing the above double integration numerically with Mathcad yields δ = 0.044 inch. In other words, the computed shrinkage of the basket is less than 0.022% of its original length.

It is concluded that for vertical configuration of storage the creep effects of the MPC basket are insignificant due to absence of any meaningful loads on the panels. Therefore, creep in the Metamic-HT fuel basket is not a matter of safety concern.

8.4.4.2 Aluminum Alloy

The basket shims are not subject to any significant loading during storage. Similar to the fuel basket, the stress levels from self-weight in long-term storage eliminates creep as a viable concern for the basket shims.

8.5 WELDING MATERIAL AND WELDING SPECIFICATION

Welds in the HI-STORM FW System are divided into two broad categories:

i. Structural welds
ii. Non-structural welds

Structural welds are those that are essential to withstand mechanical and inertial loads exerted on the component under normal storage and handling.

Non-structural welds are those that are subject to minor stress levels and are not critical to the safety function of the part. Non-structural welds are typically located in the redundant parts of the structure. The guidance in the ASME Code Section NF-1215 for secondary members may be used to determine whether the stress level in a weld qualifies it to be categorized as non-structural.

Both structural and non-structural welds must satisfy the material considerations listed in Tables 8.1.1, 8.1.2, and 8.1.3, for the MPC, the HI-STORM FW overpack and the HI-TRAC VW transfer cask, respectively. In addition, the welds must not be susceptible to any of the applicable failure modes in Table 8.1.4.

To ensure that all welds in the HI-STORM FW System shall render their intended function, the following requirements are observed:

i. The weld joint configuration is selected to accord with the function of the joint (Holtec Position Paper DS-329 [8.5.1] provided to the USNRC in Docket No. 72-1014).

ii. The welding procedure specifications comply with ASME Section IX for every Code material used in the system.

iii. The quality assurance requirements applied to the welding process correspond to the highest ITS classification of the parts being joined.

iv. The non-destructive examination of every weld is carried out using quality procedures that comply with ASME Section V.

The welding operations are performed in accordance with the requirements of codes and standards depending on the design and functional requirements of the components.

The selection of the weld wire, welding process, range of essential and non-essential variables,* and the configuration of the weld geometry has been carried out to ensure that each weld will have:

i. Greater mechanical strength than the parent metal.
ii. Acceptable ductility, toughness, and fracture resistance.
iii. Corrosion resistance properties comparable to the parent metal.
iv. No risk of crack propagation under the applicable stress levels.

The welding procedures implemented in the manufacturing of HI-STORM FW System components are intended to fulfill the above performance expectations.

Additional information on the welding for HI-STORM FW System components is provided in Section 1.2. Lists of codes and standards applicable for the manufacturing of HI-STORM FW System are also provided therein.

A list of ASME code alternatives for the MPC fabrication including welding is presented in Section 2.2. The structural strength requirements of welds including fracture toughness test requirements of weld materials are provided in Section 3.1. The confinement boundary welds and their testing requirements are discussed in Section 7.1. The inspection and testing requirements of the HI-STORM FW System component welds are provided in Section 10.1.

The weld filler material shall comply with requirements set forth in the applicable Welding Procedure Specifications qualified to ASME Section IX at the manufacturer's facility. Only those welding procedures that have been qualified to the Code are permitted in the manufacturing of HI-STORM FW components.

Review of the above shows that except for the MPC lid welds, all welds of the Enclosure Vessel are full penetration weld with volumetric NDE. All weld filler metals are specified by ASME Section II, Part C and associated AWS classification in applicable weld procedures.

The weld procedure qualification record specifies the requirements for fracture control (e.g. post weld heat treatment). The HI-STORM FW overpack and HI-TRAC VW transfer cask do not require any post weld heat treatment due to the material combinations and provisions in the applicable codes and standards. With respect to the MPC Lid-to-Shell weld, the progressive P.T. requirements on the shell/lid weld are identical to those in Docket No. 72-1014 (which are derived from the analysis summarized in Holtec Position Paper DS-213 [8.5.2], provided to the USNRC on Docket No. 72-1014.

* Please refer to Section IX of the ASME Code for the definition and delineation of essential and non-essential variables.

Non-structural welds shall meet the following requirements:

1. The welding procedure shall comply with Section IX of the ASME Code or AWS D1.1.
2. The welder shall be qualified, at minimum, to the commercial code such as ASME Section VIII, Div.1, or AWS D1.1.
3. The weld shall be visually examined by the weld operator or a Q.C. inspector qualified to Level 1 (or above) per ASNT designation.

8.6 BOLTS AND FASTENERS

Chapter 3 provides information on the structural evaluation of the bolts and fasteners. Section 3.1 discusses fracture toughness requirements for bolting materials. Section 3.3 provides the bolting materials used in the HI-STORM FW System. Section 3.3 (Table 3.3.4) provides mechanical properties of bolting materials.

Chapter 9 provides pre-tensioning requirements for HI-STORM FW System bolts to ensure that the bolts shall not be overstressed under any condition of loading applicable to the system.

Bolts and fasteners made of low alloy steel are not expected to experience any significant corrosion in the operating environment. The ISFSI operation and maintenance program shall call for coating of bolts and fasteners if the ambient environment is aggressive.

A review of the above shows that the materials for the bolts and the fasteners have been selected to possess the required tensile strengths, resistance to corrosion and brittle fracture. To prevent a change in the bolt pre-stress during operating conditions, the coefficient of thermal expansion of each bolt material has been closely matched to that of the parts being fastened together.

Preventing galling of interfacing surfaces is another critical consideration in selecting bolt materials. Use of austenitic stainless bolts on interfacing austenitic stainless steel surfaces is not permitted. All threaded surfaces are treated with a preservative to prevent corrosion. The O&M program for the storage system calls for all bolts to be monitored for corrosion damage and replaced, as necessary.

8.7 COATINGS

Protective coatings are used primarily as a corrosion barrier and/or as a means to facilitate decontamination. Coating materials for the HI-STORM FW system components are guided by the successful experience in similar service applications of the HI-STORM 100 and HI-STAR 100 components and parts. The main considerations in the selection of coatings are the ruggedness and physical integrity in the specific service environment, ease of decontamination as applicable to immersion service, thermal and radiation stability, and ease of application to facilitate touch-up activities for preventive maintenance. Surface preparation and repair are performed in accordance with manufacturer recommendations.

The coatings applied on specific HI-STORM FW System components are selected to be compatible with their respective conditions of service. For example, equipment used in the fuel pool environment must be conducive to convenient decontamination. Protective coatings are applied to surfaces vulnerable to corrosion such as exposed carbon steel surfaces on the HI-STORM FW overpack and HI-TRAC VW transfer cask. The MPC surfaces are not coated.

8.7.1 Environmental Conditions Applicable to Coating Selection and Evaluation Criteria:

8.7.1.1 Environmental Conditions

The environmental conditions that warrant consideration in the selection of coatings are:

i. Temperature, humidity, and insolation
ii. Radiation field
iii. Immersion service

Temperature, humidity, and insolation conditions may vary at different ISFSI sites. The coating selected for the HI-STORM FW overpack, which is subject to long-term exposure, must be stable under the entire range of psychometric conditions that prevail in the territorial United States. The coating selected for HI-TRAC VW must withstand the thermal exposure during fuel drying operations and during immersion in the spent fuel pool.

Stable performance under radiation is important for coatings applied on the inside surfaces of the HI-STORM FW overpack and the HI-TRAC VW transfer cask, which are proximate to the lateral surfaces of the MPC.

Immersion in the pool implies three major challenges to the coating on the HI-TRAC VW:

a. Risk of penetration of tiny contaminant particulates in the pores of the coating.
b. Chemical attack (by boric acid in PWR pools and demineralized water in BWR pools).
c. Temperature change as the transfer cask is immersed in or withdrawn from water.

Coatings that have been determined to be unsuitable for the immersion service shall not be used in the HI-TRAC VW transfer cask.

8.7.1.2 Coating Evaluation Criteria

The evaluation criteria for selecting coatings are summarized below. These criteria shall be used if a pre-approved coating listed in Subsection 8.7.2, for any reason, is no longer available for use.

	Coating Acceptance Criteria
1.	Non-reactive to the surrounding environment
2.	Structural performance (bendability, ductility, resistance to cracking, and resistance to abrasion)
3.	Adherence to base material
4.	Chemical immersion resistance, if applicable
5.	Emissivity and absorptivity consistent with thermal analysis
6.	Temperature resistance for analyzed temperature conditions with humidity and insolation, as applicable
7.	Radiation resistance for analyzed conditions

The paint suppliers may certify the properties by performance of applicable ASTM tests. In the absence of ASTM test data for a required characteristic in the above table, the coating supplier will provide evidentiary information to justify acceptance. Alternatively, Holtec International will perform its own independent tests to establish compliance with the required criteria.

8.7.2 Acceptable Coatings

Proven (previously used on HI-STORM 100 System components and other cask designs) coatings and paints that adequately satisfy the requirements are presented below and pre-approved for use on HI-STORM FW System components.

Carboguard 890 (Cycloaliphatic Amine Epoxy) of Carboline Company which demonstrates acceptable performance for short-term exposure in mild borated pool water may be used for coating the HI-TRAC VW transfer cask exterior surfaces as well as HI-STORM FW overpack surfaces. This coating is certified for immersion services and provides excellent chemical resistance and abrasion resistance. It provides a smooth surface with no porosity and thereby, excellent decontamination characteristics. No adverse interaction has been experienced in many years of use.

Thermaline 450 (Amine-Cured Novolac Epoxy) of Carboline Company may be used for coating HI-TRAC VW transfer cask internal surfaces which are exposed only to demineralized water during in-pool operations (the annulus is filled prior to placement in the spent fuel pool and the inflatable seal prevents fuel pool water in-leakage) and higher service temperatures. This coating provides excellent resistance to corrosion, abrasion, and permeation. No adverse interaction has been experienced in many years of use.

Carbozinc 11 (also known as CZ-11) may be used for coating HI-STORM FW overpack internal cavity and external surfaces (including lid surfaces). This solvent based coating material has excellent corrosion resistant properties in harsh environments and provides inorganic zinc (galvanic) protection to steel surfaces. As an alternative to the Carbozinc 11, Sherwin Williams Zinc Clad II HS, Sherwin Williams Zinc Clad II Plus may also be used.

Product information for the above coatings is provided in Appendix 8.A.

Coatings that are specified in this section shall not be substituted with another coating unless the substitute meets or exceeds the performance of the coating listed above under all the applicable coating evaluation criteria set forth in the previous subsection.

8.7.3 Coating Application

Holtec utilizes Q.A.-validated written procedures (HSP-318 [8.7.1] and HSP-319 [8.7.2]) to achieve the desired performance for the coating. These procedures provide requirements for the preparation and painting of the HI-STORM FW overpack, HI-TRAC VW transfer cask and associated components. These procedures are based on paint manufacturers' applicable specifications, instructions and recommendations.

The procedures provide details for the preparation prior to blasting, surface preparation, mixing and application, painting in the field, and touch up steps or repairs. The procedures also provide details of the dry film thickness testing and the acceptance criteria. Painting documentation is maintained for the record of the completion of various painting steps and the environmental conditions including the ambient temperature, humidity and the component surface temperature.

8.8 GAMMA AND NEUTRON SHIELDING MATERIALS

Gamma and neutron shield materials in the HI-STORM FW System are discussed in Section 1.2. The primary shielding materials used in the HI-STORM FW system, like the HI-STORM 100 system, are plain concrete, steel, lead, and water.

The plain concrete enclosed by cylindrical steel shells, a thick steel baseplate, and a top annular plate provides the main shielding function in the HI-STORM FW overpack. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation to minimize skyshine.

The transfer cask in the HI-STORM FW system (HI-TRAC VW) is provided with steel and lead shielding to ensure that the radiation and exposure objectives of 10CFR72.104 and 10CFR72.106 are met. The space between the inner shell and the middle shell is occupied by lead, conforming to ASTM B29, which provides the bulk of the cask's (gamma) radiation shielding capability. The water jacket between the middle shell and the outermost shell (filled with demineralized water or ethylene glycol fortified water, depending on the site environmental constraints) provides most of the neutron shielding capability to the cask. The water in the water jacket serves as the neutron shield on demand: When the cask is in the pool and the MPC is full of water, the water jacket is kept empty (or partially empty as necessary) to minimize the cask's weight, the neutron shielding function being provided by the water in the MPC cavity. However, when the MPC is emptied of water at the Decontamination and Assembly Station (DAS), then the neutron shielding capacity of the cask is replenished by filling the water jacket. The HI-TRAC VW bottom lid is extra thick steel to provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

8.8.1 Concrete

Appendix 1.D of HI-STORM 100 FSAR provides details of the concrete properties and the testing requirements. The *critical characteristics* of concrete are its density and compressive strength.

The density of plain concrete within the HI-STORM FW overpack is subject to a minor decrease due to long-term exposure to elevated temperatures. The reduction in density occurs primarily due to liberation of unbonded water by evaporation.

The density of concrete has been classified into three states in the published literature [8.8.1].

a) fresh density: the density of freshly mixed concrete
b) air-dry density: drying in air under ambient conditions, where moisture is lost until a quasi-equilibrium is reached
c) oven-dry density: concrete dried in an oven at 105°C (221°F)

Because the bulk temperature of concrete in HI-STORM FW is spatially variable, the oven-dry density is conservatively used as the reference density for shielding analysis.

Density loss during the initial drying process is considered in the fabrication of the HI-STORM FW overpack by providing wet concrete densities above the minimum required dry (hardened paste) density. Density loss during drying is on the order of 1% and conservatively imposes a larger delta between wet density and the minimum dry density. The data in the literature, viz., Neville [8.8.1] indicates that the density difference between the air-dry condition and oven-dry condition is about one fourth of the density difference experienced during the drying process. Therefore, the loss in density would be expected to be on the order of 0.25%. This density loss is very low and is considered too small to have a significant impact on the shielding performance of the overpack. Thus, the minimum "fresh density" during concrete placement is set equal to the reference density (Table 1.2.5) plus 1.25%.

Section 5.3 considers the minimum density requirements of concrete for effective shielding. The density requirement is confirmed per Appendix 1.D of the HI-STORM 100 FSAR.

8.8.2 Steel

Section 5.3 provides a discussion on steel as a shielding material and its composition used in the evaluation of its shielding characteristics.

8.8.3 Lead

Section 1.2 provides a discussion on lead used in HI-TRAC VW for gamma shielding. In the HI-TRAC VW transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the HI-TRAC VW bottom lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

Mechanical properties of lead are provided in Section 3.3. Section 5.3 provides the minimum density and composition (mass fraction of trace elements) of lead.

8.8.4 Water

Water is used as a neutron shield in the HI-TRAC VW transfer cask. Section 5.3 provides the minimum density requirements of water for transfer cask water jacket and inside MPC. The shielding effectiveness is calculated based on the minimum water density at the highest operating temperature. Calculations show that additives for freeze protection (at low temperature operation) such as ethylene glycol do not have any adverse effect on effectiveness of the neutron shielding function of water in the water jacket.

As discussed in Section 5.1, there is only one accident that has any significant impact on the shielding configuration. This accident is the postulated loss of the neutron shield (water) in the HI-TRAC VW. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by air.

8.9 NEUTRON ABSORBING MATERIALS

Inside the MPC enclosure vessel is a structure referred to as the fuel basket. The fuel basket is an egg-crate assemblage of Metamic-HT plates which creates prismatic cells with square cross sectional openings for fuel storage. Metamic-HT is the neutron absorber and structural material of the MPC fuel basket. Metamic-HT is a composite material of nano-particles of aluminum oxide (alumina) and finely ground boron carbide particles dispersed in a metal matrix of pure aluminum [8.9.7].

8.9.1 Qualification and Properties of Metamic-HT

The qualification and properties of Metamic-HT are presented in Appendix 1.B where its key characteristics necessary for insuring nuclear reactivity control, thermal, and structural performance are discussed. A test program configured to address the Metamic-HT properties was conducted by Holtec International and the minimum guaranteed values (MGVs) of the *critical characteristics* of Metamic-HT were determined [8.9.7] and summarized in Appendix 1.B. All testing was conducted in accordance with the applicable ASTM test standards. The role in the fuel basket safety function of each of the critical characteristics is provided in Appendix 1.B.

A rigorous quality control regimen and Holtec QA procedures ensure that all extruded Metamic-HT plates meet the requirements for the quality genre of the casks.

To ensure that the manufactured Metamic material will render its intended function with reasonable assurance, a sampling plan based on Mil Standard 105E [8.9.8] has been specified and made a part of the Metamic-HT Manufacturing Manual [8.9.6]. The Sampling plan shall provide a reasonable level of confidence that the Minimum Guaranteed Values of all critical mechanical properties will be met in the production lots. Additional information regarding manufacturing of Metamic-HT is provided in Appendix 1.B.

Chapter 2 provides discussions on criticality parameters for design basis SNF, and the controls and methods utilized for prevention of criticality.

Criticality evaluation is presented in Chapter 6. The material heterogeneity parameters are adequately characterized and controlled and the criticality calculations employ appropriate corrections when modeling the heterogeneous material as an idealized homogeneous mixture. It is demonstrated that the MPC provides criticality control for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 6.1. The effective neutron multiplication factor is limited to $k_{eff} < 0.95$ for fresh unirradiated fuel with optimum water moderation and close reflection, including all biases, uncertainties, and MPC manufacturing tolerances. Additional neutronic properties of Metamic-HT are provided in Appendix 1.B.

8.9.2 Consideration of Boron Depletion

The effectiveness of the borated neutron absorbing material used in the MPC fuel basket design requires that sufficient concentrations of boron be present to assure criticality safety during worst case design basis conditions over the design life of the MPC. Analysis discussed in Section 6.3 demonstrates that the boron depletion in the neutron absorber material is negligible over the expected service life of the HI-STORM FW System. This is due to the fact that the borated material is subjected to a relatively low neutron flux. Analyses show that the depletion of boron is a small fraction of the quantity present. Therefore, sufficient levels of boron will remain in the fuel basket neutron absorbing material to maintain criticality safety functions over the design life of the MPC. Furthermore, the boron content of Metamic-HT used in the criticality safety analysis is conservatively based on the minimum specified boron areal density (rather than the nominal), which is further reduced by 10% (see Chapter 6) for conservatism in the analysis.

8.10 CONCRETE AND REINFORCING STEEL

The HI-STORM FW System does not utilize concrete with rebar. The plain concrete used in the HI-STORM FW overpack serves as the neutron shielding. The absence of rebar in the HI-STORM FW overpack concrete ensures that radiation streaming paths due to the development of cracks and discontinuities at the rebar/concrete interfaces will not develop. Concrete in the overpack is not considered as a structural member, except to withstand compressive, bearing, and penetrant loads. Therefore the mechanical behavior of concrete must be quantified to determine the stresses in the structural members (steel shells surrounding it) under accident conditions.

Section 3.3 provides the concrete mechanical properties. Allowable, bearing strength in concrete for normal loading conditions is calculated in accordance with ACI 318-05 [8.3.2]. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. Appendix 1.D in the HI-STORM 100 FSAR provides additional information on the requirements on plain concrete for use in HI-STORM FW storage overpack.

To enhance the shielding performance of the HI-STORM FW storage overpack, high density concrete can be used during fabrication. The permissible range of concrete densities is specified in Table 1.2.5.

Review of the above shows that the HI-STORM FW System concrete components are acceptable. All concrete is either encased in steel or covered underneath the overpack lid, therefore; it is not subject to weathering or other atmospheric degradation, even in marine environments. To ensure that the concrete performs its primary function (shielding integrity/effectiveness) tests are performed as required by Chapter 10.

8.11 SEALS

The HI-STORM FW System does not rely upon mechanical seals for maintaining the integrity of the Confinement Boundary. The MPC Vent/Drain caps washers are made of a soft and malleable metal such as aluminum 1100.

The HI-TRAC VW transfer cask bottom lid utilizes a gasket to prevent ingress of pool water when the cask is staged in the fuel pool and leakage during MPC processing operations. Gaskets used may be silicone, neoprene, and a similar elastomeric material that is inert in the pool's aqueous environment.

In selecting the gasket material, it is necessary to ensure that none of the following materials will leach out in the pool water in measurable quantities.

- Viton
- Saran
- Silastic L8-53
- Teflon
- Nylon
- Carbon steel
- Neoprene or similar materials made of halogen containing elastomers
- Rubber bonded asbestos
- Polyethelene film colored with pigments over 50 ppm fluorine, measurable amount of mercury or halogens, or more than 0.05% lead
- Materials containing lead, mercury, sulfur, phosphorus, zinc, copper and copper alloys, cadmium, tin, antimony, bismuth, mischmetal, magnesium oxide, and halogens exceeding 75 ppm (including cleaning compound).

The gaskets used in the HI-TRAC VW shall be the same or equivalent to those that have proven to be satisfactory in prior service (such as in other Holtec transfer casks).

The mechanical design details of the gasketed joint in the transfer cask follow the guidelines in Chapter 3 of [8.11.1], which recommend joints subjected to cyclic loadings to be made of the "controlled compression" genré. The "controlled compression" joint minimizes cyclic damage to the gasket.

The O&M program for the storage system calls for HI-TRAC VW transfer cask elastomeric seals to be inspected for damage and replaced on an appropriate schedule as recommended by the manufacturer.

8.12 CHEMICAL AND GALVANIC REACTIONS

The materials used in the HI-STORM FW System are examined to establish that these materials do not participate in any chemical or galvanic reactions when exposed to the various environments during all normal operating conditions and off-normal and accident events.

The following acceptance criteria for chemical and galvanic reactions are extracted from ISG-15 [8.1.1] for use in HI-STORM FW components.

a. The DCSS should prevent the spread of radioactive material and maintain safety control functions using, as appropriate, noncombustible and heat resistant materials.

b. A review of the DCSS, its components, and operating environments (wet or dry) should confirm that no operation (e.g., short term loading/unloading or long-term storage) will produce adverse chemical and/or galvanic reactions, which could impact the safe use of the storage cask.

c. Components of the DCSS should not react with one another, or with the cover gas or spent fuel, in a manner that may adversely affect safety. Additionally, corrosion of components inside the containment vessel should be effectively prevented.

d. The operating procedures should ensure that no ignition of hydrogen gas should occur during cask loading or unloading.

e. Potential problems from general corrosion, pitting, stress corrosion cracking, or other types of corrosion, should be evaluated for the environmental conditions and dynamic loading effects that are specific to the component.

The materials and their ITS pedigree are listed in the drawing package provided in Section 1.5. The compatibility of the selected materials with the operating environment and to each other for potential galvanic reactions is discussed in this section.

8.12.1 Operating Environments

During fuel loading, handling or storage the components of the HI-STORM FW System experience the following environments (see Tables 8.1.1, 8.1.2, and 8.1.3).

- Spent Fuel Pool Water – During the fuel loading steps, the MPC confinement space is flooded with water (borated water in PWRs and demineralized water in BWRs). As water is withdrawn from the MPC space, the temperature of its contents rises, facilitating an Arrhenius-like acceleration of any chemical reaction that may occur in the presence of water and water vapor or boric acid (in PWRs). These same conditions would exist in the event an MPC needs to be unloaded and the MPC is reflooded prior to lid removal.

- Helium – During loading operations, all water is removed from the interior of the MPC and an inert gas is injected. Internal MPC components get exposed to dry helium under pressure during storage.

- External atmosphere – During long term storage the casks are exposed to outside atmosphere, air with temperature variations, solar radiation, rain, snow, ice, etc.

As discussed below, the components of the HI-STORM FW System has been engineered to ensure that the environmental conditions expected to exist at nuclear power plant installations do not prevent the cask components from rendering their respective intended functions.

8.12.2 Compatibility of MPC Materials

8.12.2.1 MPC Confinement Boundary Materials

Austenitic Stainless Steels

The MPC confinement boundary is composed entirely of corrosion-resistant austenitic stainless steel. The corrosion-resistant characteristics of such materials for dry SNF storage canister applications, as well as the protection offered by these materials against other material degradation effects, are well established in the nuclear industry. The available austenitic stainless steels are AISI Types 304, 304LN, 316 and 316LN containing a minimum of 16% chromium and 8% nickel, and at least traces of molybdenum. The passive films (formed due to atmospheric exposure) of stainless steels range between 10 to 50 angstroms ($1x10^{-6}$ to $5x10^{-6}$ mm) thick [8.12.4]. Of all types of stainless steels (i.e., austenitic, ferritic, martensitic, precipitation hardenable and two-phase), "the austenitic stainless alloys are considered the most resistant to industrial atmospheres and acid media" [8.12.4].

The MPC contains no gasketed, threaded, or packed joints for maintaining confinement. The all-welded construction of the MPC confinement boundary and the inert backfill gas within ensures that the interior surfaces and the MPC internals (Metamic-HT baskets, shims, etc.) are not subject to corrosion. Exterior MPC surfaces would be exposed to the ambient environment while inside of a HI-STORM FW storage overpack or a HI-TRAC VW transfer cask.

Austenitic Stainless Steels in Demineralized and Borated Water Environments

The average MPC may be in contact with borated and/or demineralized water at temperatures below boiling and at pressures of up to three atmospheres (not including hydrotest) for approximately 2 to 3 days. For PWRs, the soluble boron levels are typically maintained at or below 2,500 ppm (0.25% boric acid solution). Experimental corrosion data for AISI Type 304 and 316 stainless steels (Swedish Designations SIS-14-2333 and SIS-14-2343, respectively) are available from the Swedish Avesta Jernverk laboratory [8.12.4]. Corrosive media evaluated in these tests include 4% (40,000 ppm) and 20% (200,000 ppm) boric acid solutions and water, all at boiling. Under the evaluated conditions, the tested steels are identified as "fully resistant", with corrosion rates of less than 0.1 mm per year. Even more extensive experimental corrosion data is available from ASM International [8.12.1]. For test conditions without rapid agitation, similar to conditions that would exist during MPC fuel loading in a spent fuel pool, all austenitic stainless steels available for MPC fabrication (i.e., AISI Types 304, 304LN, 316 and 316LN) are extremely resistant to corrosion in boric acid and water. More specifically, one set of data (UNS No. S30400) for 2.5% boric acid solution and water at 90.6°C (195°F), under no aeration and rapid agitation yielded a maximum corrosion rate of 0.003 mm per year [8.12.1].

No structural effects from any potential corrosion from demineralized and borated water environments are expected. Loading of a dry storage cask with reasonable delays can take up to two weeks. Adjusting the worst-case data for a 0.25% boric acid concentration the maximum thinning of any structural member in an MPC is only 4.80 x 10^{-6} mm (1.89 microinches). This is a negligibly small fraction (0.0006%) of the thickness of the thinnest structural member 7.9 mm (0.3125 in.) and a negligibly small fraction (0.004%) of the tolerance on the material thickness (0.045 in.) permitted by the governing ASME Code [8.12.2].

Austenitic Stainless Steels and Crud

Corrosion products cause "crud" deposits on fuel assemblies. Industry experience shows that crud, which is stable in oxygenated solutions, has not been found to contain materials that can react with stainless steel and cause significant degradation. Crud may leave a slight film of rust on the interior surfaces of the MPC during fuel loading and closure activities.

Austenitic Stainless Steels and Boron Crystals

Dry boron or boric acid crystals that remain in the MPC after drying and helium backfill are expected to have negligible corrosive effects on stainless steel due to the absence of the necessary reagents (oxygen and moisture).

Austenitic Stainless Steels and Marine Environments

The MPC is designed to be loaded with spent fuel assemblies from most light water reactor (LWR) nuclear power plants. LWR nuclear power plants, in general, are located near large bodies of water to ensure an adequate supply of cooling water. As a result many nuclear power plants and, subsequently, many potential ISFSI sites are located in coastal areas where dissolved salts may be present in atmospheric moisture. Casks deployed at coastal ISFSI sites that would be exposed to the harsh marine environment for prolonged periods must not suffer corrosion that will impair their functionality.

Extensive data show corrosion rates (pitting) to 0.0018 (mm/yr) for 304, 304LN, 316 and 316LN in marine environments at ambient temperatures after 26 years [8.12.1]. Using this bounding corrosion rate data, a Holtec Position Paper [8.12.3] estimates the total corrosion of the external surface of the MPC in 100 years of service is about half a millimeter which is significantly smaller than the available design margins in the material thickness. It is to be noted that this upper-bound is estimated for an extreme hypothetical marine environment. As discussed earlier for inland applications the corrosion rates are insignificant.

Therefore, corrosion of the MPC in long-term storage is not a credible safety concern.

Austenitic Stainless Steels and Hydrogen Damage

Traces of hydrogen may be present under the MPC Lid during welding operations. The hydrogen content is limited due to a low hydrogen generation rate and the (required) purging of the underside of the lid with helium. Hydrogen damage is classified into four distinct types (1) hydrogen blistering, (2) hydrogen embrittlement, (3) decarburization, (4) hydrogen attack. Decarburization and hydrogen attack are high temperature processes and therefore may be of concern during cooling of the weld puddle. Austenitic stainless steels are one of the few metals that perform satisfactorily at all temperatures and pressures in the presence of hydrogen [8.12.6]. Considering the limited hydrogen concentration, limited time (2-3 days) for fuel loading and limited pressures and temperatures (with the exception of high temperatures at the lid to shell weld), hydrogen damage is not an applicable corrosion mechanism during fuel loading. With respect to the lid to shell weld, the weld design, use of a continuous inert gas purge, the weld method and NDE inspections provide assurance that the weld has no credible damage and is of high integrity.

8.12.2.2 Materials of MPC Internals

The internals of the MPC consists of Metamic-HT fuel baskets and aluminum alloy shims for basket support. Besides these internals, SNF, possible failed fuel and/or damaged fuel with containers, and non-fuel hardware, a sealed MPC may also contain boric acid crystals (in PWRs) and cruds. The cleanliness requirements and inspections during fabrication and fuel loading operations ensure that the MPC has minimal surface debris and impurities.

Tests on Metamic-HT

Extensive tests [8.9.7] have been conducted to establish material properties of Metamic-HT including its corrosion-resistance characteristics. The Metamic-HT specimens were used for corrosion testing in demineralized water and in 2000 ppm boric acid solution. The tests concluded that the Metamic-HT panels will sustain no discernible degradation due to corrosion when subjected to the severe thermal and aqueous environment that exists around a fuel basket during fuel loading or unloading conditions.

Aluminum Alloy

Aluminum alloy used in the fuel basket shims are hard anodized. The anodizing is an electrolytic passivation process used to increase the thickness of the natural oxide layer on the surface of metal parts. Anodizing increases corrosion resistance and wear resistance of the material surface. There is no mechanistic process for the basket shims with hard anodized surface to react with borated water or demineralized water during fuel loading operation. Under the long-term storage condition, the basket shims are exposed to dry and inert helium with no potential for reaction.

Effect of Forced Helium Dehydration (FHD) Process

The operation of the FHD consists of flowing hot dry helium through the MPC at pressures and temperaturelimited by the MPC design pressure and temperature of the MPC. Due to the purity of the helium stream and the relatively short duration (normally 10 to 60 hours), no significant corrosion mechanisms are identified.

Maintenance of Helium Atmosphere

The inert helium atmosphere in the MPC provides a non-oxidizing environment for the SNF cladding to assure its integrity during long-term storage. The preservation of the helium atmosphere in the MPC is assured by the robust design of the MPC Confinement Boundary (see Section 7.1). Maintaining an inert environment in the MPC mitigates conditions that might otherwise lead to SNF cladding failures. The required mass quantity of helium backfilled into the canister at the time of closure and the associated fabrication and closure requirements for the canister are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout the MPC's service life.

Allowable Fuel Cladding Temperatures

The helium atmosphere in the MPC promotes heat removal and thus reduces SNF cladding temperatures during dry storage. In addition, the SNF decay heat will substantially attenuate over the dry storage period. Maintaining the fuel cladding temperatures below allowable levels during long-term dry storage mitigates the damage mechanism that might otherwise lead to SNF cladding failures. The allowable long-term SNF cladding temperatures used for thermal acceptance of the MPC design are conservatively determined, as discussed in Section 4.3.

8.12.2.3 Galvanic Corrosion

The MPC is principally constructed of stainless steel shell and Metamic-HT. Borated aluminum and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from Metamic (classic) and stainless steel materials. Not one case of chemical or galvanic degradation has been found in such fuel racks. This experience provides a sound basis to conclude that corrosion will not occur in these materials. For further protection, both Metamic-HT and aluminum basket shims are installed in the anodized state in the MPC.

Furthermore, galvanic corrosion is not an applicable mechanism since the interior of the MPC during normal operation is essentially devoid of any moisture and the MPC shell surfaces are expected to be practically free from condensation. Finally, the interior of the carbon steel HI-STORM FW overpack is painted to inhibit corrosion.

During long-term storage in the HI-STORM FW overpack, the MPC operates at elevated temperatures under normal conditions while inside the HI-STORM. The external ambient environment normally consists of atmospheric conditions, which include humidity and perhaps airborne contaminants such as sulfur dioxide, chlorine gas, sulfur gas and ozone. The interior is backfilled with highly pure helium. The spent fuel irradiates the MPC but at much lower levels than those experienced in an operating reactor. It is recognized that in general the higher the temperature the higher the rate of chemical reaction. It is also recognized that moisture will not exist on the MPC exterior surfaces for many years since moisture will not condense on hot surfaces and the protection afforded by the HI-STORM FW overpack. It is estimated that it would take decades for the hottest MPC to approach ambient temperatures and once at ambient temperature, any MPC surfaces will be highly corrosion resistance even when wet.

8.12.2.4 Cyclic Fatigue

As discussed in Section 3.1, passive non-cyclic nature of dry storage conditions does not subject the MPC to conditions that might lead to structural fatigue failure. Ambient temperature and insolation cycling during normal dry storage conditions and the resulting fluctuations in MPC thermal gradients and internal pressure is the only mechanism for fatigue. These low-stress, high-cycle conditions cannot lead to a fatigue failure of the MPC that is made from stainless alloy stock (endurance limit well in excess of 20,000 psi). All other off-normal or postulated accident conditions are infrequent or one-time occurrences, which cannot produce fatigue failures.

8.12.3 Compatibility of HI-STORM FW Overpack Materials

The principal operational considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

Exposure to Environmental Effects

All exposed surfaces of the HI-STORM FW overpack are made from ferritic steels that are readily painted. Concrete, which serves strictly as a shielding material, is encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete, are ruled out for HI-STORM FW overpack. Under normal storage conditions, the bulk temperature of the HI-STORM FW overpack will change very gradually with time because of its large thermal inertia. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM FW overpack. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM FW because HI-STORM FW does not rely on rebars (indeed, it contains no rebars). As discussed in Appendix 1.D of the HI-STORM 100 FSAR, the aggregates, cement and water used in the storage cask concrete are adequately controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation. In addition, the storage overpack is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the service life of the storage overpack as catalogued in Section 2.2 and evaluated in Chapter 11.

Material Degradation

The relatively low neutron flux to which the storage overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The ambient environment of the ISFSI storage pad mitigates damage due to exposure to corrosive and aggressive chemicals that may be produced at other industrial plants in the surrounding area.

Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage overpack throughout its service life are defined in Section 10.2. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STORM FW system is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [8.12.5], which concluded that dry storage systems designed, fabricated, inspected, and operate in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

8.12.4 Compatibility of HI-TRAC VW Transfer Cask Materials

The principal design considerations that bear on the adequacy of the HI-TRAC VW Transfer Cask for the service life are addressed as follows:

Exposure to Environmental Effects

All transfer cask materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The HI-TRAC VW is designed for repeated normal condition handling operations with a high factor of safety to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the cask's materials, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading, as discussed in Section 8.4 in the foregoing.

Material Degradation

All transfer cask materials that are susceptible to corrosion are coated. The controlled environment in which the HI-TRAC VW is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the HI-TRAC VW materials are subjected do not result in radiation embrittlement or degradation of the shielding materials in the HI-TRAC VW that could impair the intended safety function. The HI-TRAC VW transfer cask materials have been selected for durability and wear resistance for their deployment.

Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the HI-TRAC VW transfer cask throughout its service life are defined in Section 10.2. These requirements include provisions for routine inspection of the HI-TRAC VW transfer cask for damage prior to each use. Precautions are taken during bottom lid handling operations to protect the sealing surfaces of the bottom lid. The leak tightness of the liquid neutron shield is verified periodically. The water jacket pressure relief devices and connections for water injection/removal have been engineered for convenient removal and replacement.

8.12.5 Potential Combustible Gas Generation

To ensure safe fuel loading operation the operating procedure described in Chapter 9 provides for the monitoring of hydrogen gas in the area around the MPC lid prior to and during welding or cutting activities. Although the aluminum surfaces (Metamic-HT basket and aluminum basket shims) are anodized, there is still a potential for generation of hydrogen in minute amounts when immersed in spent fuel pool water for an extended period. Accordingly, as a defense-in-depth measure, the lid welding procedure requires purging the space below the MPC lid prior to and during welding or cutting operation to eliminate any potential for formation of any combustible mixture of hydrogen and oxygen. Following the completion of the MPC lid welding and hydrostatic testing the MPC is drained and dried. As discussed earlier, after the completion of the drying operation there is no credible mechanism for any combustible gases to be generated within the MPC.

8.12.6 Oxidation of Fuel During Loading/Unloading Operations

During the loading and unloading operations in a spent fuel pool, the fuel cladding is surrounded by water. During fuel drying operation the water is displaced with a non-oxidizing gas environment. Therefore, there is no credible mechanism for oxidation of fuel.

8.12.7 Conclusion

The above discussion leads to the conclusion that the materials selected for the HI-STORM FW System components are compatible with the environment for all operating conditions. There is no potential for significant corrosion, chemical reaction or galvanic reaction to shorten the intended service life of the equipment. In other words, the acceptance criteria set forth in ISG-15 are completely satisfied.

8.13 FUEL CLADDING INTEGRITY

8.13.1 Regulatory Guidance

The acceptance criteria from ISG-11that apply to the fuel cladding are:

a. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad).

   However, for low burnup fuel, a higher short-term temperature limit may be used, if it can be shown by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13.053 psi) for the temperature limit proposed.

b. During loading operations, for high burnup fuel, repeated thermal cycling (repeated heatup/cooldown cycles) may occur but should be limited to less than 10 cycles, with cladding temperature variations that are less than 65°C (149°F) each.

c. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

The ISG-15 guidance on cladding integrity in its entirety provides the following supplemental requirements:

a. The cladding temperature should be maintained below maximum allowable limits, and an inert environment should be maintained inside the cask cavity to maintain reasonable assurance that the spent fuel cladding will be protected against degradation that may lead to gross rupture, loss of retrievability, or severe degradation.

b. Cladding should not rupture during re-flood operations.

8.13.2 Measures to Meet Regulatory Guidance

The HI-STORM FW System features and processes minimize the potential for any spent fuel cladding degradation during transfer and storage conditions by limiting the fuel cladding temperature and the environment around the fuel rod to within ISG-11 limits (Table 4.3.1).

The highly pure helium under positive pressure in the canister limits the amount of oxidants and controls the cladding temperature. The MPC drying and helium backfilling operations result in the creation of an inert environment around the fuel. As prescribed by NUREG-1536 [8.3.3], if the classical vacuum drying method is used, the partial pressure of water vapor is brought down to below 3 torr to minimize [8.13.1] residual oxidizing gas concentration.

An alternative method (preferred) for drying the MPC internals utilizes Holtec's patented Forced Helium Dehydration technology [8.13.1, 8.13.2] described in the HI-STORM 100 FSAR (Appendix 2.B). The Forced Helium Dehydrator has been successfully used at numerous nuclear plants since its regulatory approval in 2001. The efficacy of the Forced Gas Dehydrator (FGD) has been tested in a full-scale demonstration [8.13.4] for demoisturizing simulated water-logged RBMK fuel [8.13.3].

The FHD uses helium as the working substance. The use of the FHD prevents the elevation of the fuel cladding temperature during drying, which is a chief demerit of the vacuum drying method. The use of the FHD method of drying is compulsory for high burnup fuel to protect its (relatively) ductility challenged cladding from severe thermal transients.

Chapter 2 provides the allowable fuel cladding temperature limits along with other design conditions. Chapter 4 presents performance evaluation of the HI-STORM FW System under normal conditions of storage, MPC temperatures during moisture removal operations and HI-STORM FW System long term storage maximum temperature conditions. Chapter 4 provides MPC temperatures under various accident conditions. It is demonstrated that the maximum calculated fuel cladding temperature is within 400°C (752°F) with substantial margins for normal conditions of storage and short-term loading operations. For off-normal and accident conditions, the maximum cladding temperature does not exceed 570°C (1058°F).

The short-term operations described in Chapter 9 are specifically configured to prevent severe thermal stresses in the fuel cladding due to rapid thermal transients.

The thermal stresses from MPC reflood analysis during fuel unloading operations shall be lower than typical MPCs because the HI-STORM FW fuel assemblies operate at considerably lower temperatures at Design Basis heat loads (see Chapter 4) than is permitted by ISG-11.

8.14 EXAMINATION AND TESTING

Examination and testing are integral parts of manufacturing of the HI-STORM FW System components. A comprehensive discussion on the examinations and testing that are conducted during the manufacturing process is provided in Section 10.1. The applicable codes and standards used are also referred and the acceptance criteria are listed.

8.14.1 Helium Leak Testing of Canister Welds

Helium leakage testing of the MPC shop welds (shell seams and shell-to-baseplate shop welds) shall be performed in accordance with the leakage test methods and procedures of ANSI N14.5 [8.14.1]. Acceptance criterion is specified in Chapter 10. Testing shall be performed in accordance with written and approved procedures.

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

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 criteria are met.

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are not required.

Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in Chapter 9 of this SAR and the acceptance criteria are defined in the Technical Specifications for the HI-STORM FW System.

8.14.2 Periodic Inspections

Post-fabrication inspections are discussed in Section 10.2 as part of the HI-STORM FW System maintenance program. Inspections are conducted prior to fuel loading or prior to each fuel handling campaign. Other periodic inspections are conducted during storage.

The HI-STORM FW overpack is a passive device with no moving parts. Overpack vent screens are inspected monthly for damage, holes, etc. The overpack external surface including identification markings is visually examined annually. The temperature monitoring system, if used, is inspected per licensee's QA program and manufacturer's recommendations. HI-TRAC VW transfer cask visual inspection is performed annually for compliance with the licensing drawings.

8.15 CONCLUSION

The preceding sections describe the materials used in important to safety SSCs and the suitability of those materials for their intended functions in the HI-STORM FW System.

The requirements of 10CFR72.122(a) are met: The material properties of SSCs important to safety conform to quality standards commensurate with their safety functions.

The requirements of 10CFR72.104(a), 106(b), 124, and 128(a)(2) are met: Materials used for criticality control and shielding are adequately designed and specified to perform their intended function.

The requirements of 10CFR72.122(h)(1) and 236(h) are met: The design of the DCSS and the selection of materials adequately protect the spent fuel cladding against degradation that might otherwise lead to gross rupture of the cladding.

The requirements of 10CFR72.236(h) and 236(m) are met: The material properties of SSCs important to safety will be maintained during normal, off-normal, and accident conditions of operation as well as short-term operations so the spent fuel or MPC, as appropriate, can be readily retrieved without posing operational safety problems.

The requirements of 10CFR72.236(g) are met: The material properties of SSCs important to safety will be maintained during all conditions of operation so the spent fuel can be safely stored for the specified service life and maintenance can be conducted as required.

The requirements of 10CFR72.236(h) are met: The HI-STORM FW System employs materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials should not degrade over time or react with one another during long-term storage.

8.16 REFERENCES

[8.1.1] ISG-15, "Materials Evaluation," U.S. Nuclear Regulatory Commission, Washington, DC, Revision 0, January 2001.

[8.1.2] ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," U.S. Regulatory Commission, Washington, DC, November 2003.

[8.3.1] ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, New York, NY, (2007).

[8.3.2] ACI 318-2005, "Building Code Requirements for Structural Concrete," American Concrete Institute, Ann Arbor, MI.

[8.3.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," U.S. Nuclear Regulatory Commission, Washington, DC, January 1997.

[8.4.1] ASME Boiler & Pressure Vessel Code, Section III, Part D, 2007 Edition.

[8.5.1] Holtec Position Paper DS-329, "Stress Limits, Weld Categories, and Service Conditions", (Holtec Proprietary).

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APPENDIX 8.A

Datasheets for Coatings and Paints[§]

[§] The materials in this Appendix can also be found in the suppliers' website.

product data

carboline®

Carboguard® 890 & 890 LT

Selection & Specification Data

Generic Type

Cycloaliphatic Amine Epoxy

Description

Highly chemical resistant epoxy mastic coating with exceptionally versatile uses in all industrial markets. Self-priming and suitable for application over most existing coatings, and tightly adherent to rust. Carboguard 890 serves as stand-alone system for a variety of chemical environments. Carboguard 890 is also designed for various immersion conditions.

Features

- Excellent chemical resistance
- Surface tolerant characteristics
- Conventional and low-temperature versions
- Self-priming and primer/finish capabilities
- Very good abrasion resistance
- VOC compliant to current AIM regulations
- Suitable for use in USDA inspected facilities

Color

Refer to Carboline Color Guide. Certain colors may require multiple coats for hiding. Note: The low temperature formulation will cause most colors to yellow or discolor more than normal in a short period of time. (Epoxies lose gloss, discolor and chalk in sunlight exposure.)

Finish

Gloss

Primers

Self-priming. May be applied over inorganic zinc primers and other tightly adhering coatings. A mist coat may be required to minimize bubbling over inorganic zinc primers.

Topcoats

Acrylics, Epoxies, Polyurethanes

Dry Film Thickness

4.0-6.0 mils (100-150 microns) per coat
6.0-8.0 mils (150-200 microns) over light rust and for uniform gloss over inorganic zincs.
Don't exceed 10 mils (250 microns) in a single coat. Excessive film thickness over inorganic zincs may increase damage during shipping or erection.

Solids Content

By Volume (890): 75% ± 2%
(890LT): 80% ± 2%

Theoretical Coverage Rate

890: 1203 mil ft² (30.0 m²/l at 25 microns)
241 ft² at 5 mils (6.0 m²/l at 125 microns)
890LT: 1283 mil ft² (31.0 m²/l at 25 microns)
257 ft² at 5 mils (6.3 m²/l at 125 microns)
Allow for loss in mixing and application

VOC Values

	890	890 LT
As supplied	1.7lbs/gal (214 g/l)	1.5lbs/gal (180g/l)
Thinned w/#2*:	7oz/gal=2.0lbs/gal (250g/l) 13oz/gal=2.2lbs/gal (271g/l)	15oz/gal=2.0lbs/gal (250g/l)
Thinned w/#33*:	7oz/gal=2.0lbs/gal (250g/l) 16oz/gal=2.3lbs/gal (285g/l)	14oz/gal=2.0 lbs/gal (250g/l) 16oz/gal=2.1lbs/gal (258g/l)

*Use Thinner #76 up to 8 oz/gal for 890 and 16 oz/gal for 890 LT where non-photochemically reactive solvents are required.

Dry Temp. Resistance

Continuous: 250°F (121°C)
Non-Continuous: 300°F (149°C)
Discoloration and loss of gloss is observed above 200°F (93°C).

Limitations

Do not apply over latex coatings. For immersion projects use only factory made material in special colors. Consult Technical Service for specifics. Carboguard 890 LT should not be used for immersion and should only be used as a primer or intermediate coat. Discoloration may be objectionable if used as a topcoat.

Substrates & Surface Preparation

General

Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.

Steel

Immersion: SSPC-SP10
Non-immersion: SSPC-SP6
1.5-3.0 mils (38-75 microns)
SSPC-SP2 or SP3 are suitable cleaning methods for mild environments.

Galvanized Steel

Prime with specific Carboline primers as recommended by your Carboline Sales Representative. Refer to the specific primer's Product Data Sheet for substrate preparation requirements.

Concrete or CMU

Concrete must be cured 28 days at 75°F (24°C) and 50% relative humidity or equivalent. Prepare surfaces in accordance with ASTM D4258 Surface Cleaning of Concrete and ASTM D4259 Abrading Concrete. Voids in concrete may require surfacing. Mortar joints should be cured a min of 15 days. Prime with itself, Carboguard® 1340, or suitable filler/sealer.

Drywall & Plaster

Joint compound and plaster should be fully cured prior to coating application. Prime with Carbocrylic® 120 or Carboguard 1340.

Previously Painted Surfaces

Lightly sand or abrade to roughen surface and degloss the surface. Existing paint must attain a minimum 3B rating in accordance with ASTM D3359 "X-Scribe" adhesion test.

Performance Data

Test Method	System	Results	Report #
ASTM D3359 Adhesion	Blasted Steel 1 ct. 890	5A	0270
ASTM D4060 Abrasion	Blasted Steel 1 ct. Epoxy Pr. 1 ct. 890	85 mg. loss after 1000 cycles, CS17 wheel, 1000 gm. load	02411
ASTM B117 Salt Fog	Blasted Steel 2 cts. 890	No effect on plane, rust in scribe. 1/16" undercutting at scribe after 2000 hours	02594
ASTM B117 Salt Fog	Blasted Steel 1 ct. IOZ 1 ct. 890	No effect on plane, no rust in scribe and no undercutting after 4000 hours	L40-42,45,95
ASTM D1735 Water Fog	Blasted Steel 1 ct. Epoxy Pr. 1 ct. 890	No blistering, rusting or delamination after 2800 hours	08564
ASTM D3363 Pencil Hardness	Blasted Steel 2 cts. 890	Greater than 8H	02775
ASTM D2486 Scrub Resistance	Blasted Steel 1 ct. 890	93% gloss retained after 10,000 cycles w/ liquid scrub medium	03142
ASTM E84 Flame and Smoke	2 ct. 890	5 Flame 5 Smoke Class A	03110

Test reports and additional data available upon written request.

April 2007 replaces February 2007

0986/0983

Carboguard® 890 & 890 LT

Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modifications to these guidelines to achieve the desired results. **General Guidelines:**

Spray Application (General) This is a high solids coating and may require adjustments in spray techniques. Wet film thickness is easily and quickly achieved. The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

Conventional Spray Pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, .070" I.D. fluid tip and appropriate air cap.

Airless Spray

Pump Ratio:	30:1 (min.)*
GPM Output:	3.0 (min.)
Material Hose:	3/8" I.D. (min.)
Tip Size:	.017"-.021"
Output PSI:	2100-2300
Filter Size:	60 mesh

*Teflon packings are recommended and available from the pump manufacturer.

Brush & Roller (General) Multiple coats may be required to obtain desired appearance, recommended dry film thickness and adequate hiding. Avoid excessive re-brushing or re-rolling. For best results, tie-in within 10 minutes at 75°F (24°C).

Brush Use a medium bristle brush.

Roller Use a short-nap synthetic roller cover with phenolic core.

Mixing & Thinning

Mixing Power mix separately, then combine and power mix. DO NOT MIX PARTIAL KITS.

Ratio 890 and 890 LT 1:1 Ratio (A to B)

Thinning*
Spray: Up to 13 oz/gal (10%) w/ #2
Brush: Up to 16 oz/gal (12%) w/ #33
Roller: Up to 16 oz/gal (12%) w/ #33
Thinner #33 can be used for spray in hot/windy conditions. Use of thinners other than those supplied or recommended by Carboline may adversely affect product performance and void product warranty, whether expressed or implied.
*See VOC values for thinning limits.

Pot Life
890 3 Hours at 75°F (24°C)
890 LT 2 Hours at 75°F (24°C)
Pot life ends when coating loses body and begins to sag. Pot life times will be less at higher temperatures.

Cleanup & Safety

Cleanup Use Thinner #2 or Acetone. In case of spillage, absorb and dispose of in accordance with local applicable regulations.

Safety Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.

Ventilation When used as a tank lining or in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. User should test and monitor exposure levels to insure all personnel are below guidelines. If not sure or if not able to monitor levels, use MSHA/NIOSH approved supplied air respirator.

Caution This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

Application Conditions

890

Condition	Material	Surface	Ambient	Humidity
Normal	60°-85°F (16°-29°C)	60°-85°F (16°-29°C)	60°-90°F (16°-32°C)	0-80%
Minimum	50°F (10°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	125°F (52°C)	110°F (43°C)	80%

890 LT

Condition	Material	Surface	Ambient	Humidity
Normal	60-85°F (16-29°C)	60-85°F (16-29°C)	60-90°F (16-32°C)	10-80%
Minimum	40°F (4°C)	35°F (2°C)	35°F (2°C)	0%
Maximum	90°F (32°C)	125°F (52°C)	110°F (43°C)	80%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

Curing Schedule

890 (Based on 4–8 mils, 100-200 microns dry film thickness.)

Surface Temp. & 50% Relative Humidity	Dry to Recoat	Dry to Topcoat w/ Other Finishes	Final Cure	
			General	Immersion
50°F (10°C)	12 Hours	24 Hours	3 Days	N/R
60°F (16°C)	8 Hours	16 Hours	2 Days	10 Days
75°F (24°C)	4 Hours	8 Hours	1 Day	5 Days
90°F (32°C)	2 Hours	4 Hours	16 Hours	3 Days

890 LT (Based on 5 mils, 125 microns dry film thickness.)

Surface Temp. & 50% Relative Humidity	Dry to Touch	Dry to Handle	Dry to Recoat & Topcoat w/ Others	Final Cure General Service
35°F (2°C)	5 Hours	18 Hours	20 Hours	7 Days
40°F (4°C)	4.5 Hours	15.5 Hours	16 Hours	5 Days
50°F (10°C)	3.5Hours	6.5 Hours	12 Hours	3 Days
60°F (16°C)	2 Hours	5 Hours	8 Hours	2 Days
75°F (24°C)	1.5Hours	2 Hours	4 Hours	24 Hours
90°F (32°C)	1 Hour	1.5 Hours	2 Hours	16 Hours

Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Excessive humidity or condensation on the surface during curing can interfere with the cure, can cause discoloration and may result in a surface haze. Any haze or blush must be removed by water washing before recoating. During high humidity conditions, it is recommended that the application be done while temperatures are increasing. **Maximum recoat/topcoat times are 30 days for epoxies and 90 days for polyurethanes at 75°F (24°C).** If the maximum recoat times have been exceeded, the surface must be abraded by sweep blasting or sanding prior to the application of additional coats. 890 LT applied below 50°F (10°C) may temporarily soften as temperatures rise to 60°F (16°C). This is a normal condition and will not affect performance.

Packaging, Handling & Storage

Shipping Weight (Approximate)	2 Gallon Kit 29 lbs (13 kg)	10 Gallon Kit 145 lbs (66 kg)
Flash Point (Setaflash)	89°F (32°C) for Part A; 890 & 890 LT 73°F (23°C) for Part B; 890 & 890 LT	
Storage Temperature & Humidity	40° -110°F (4°-43°C) Store indoors. 0-100% Relative Humidity	
Shelf Life: 890 & 890LT	Part A: Min. 36 months at 75°F (24°C) 890 Part B: Min. 15 months at 75°F (24°C) 890 LT Part B: Min. 15 months at 75°F (24°C)	

***Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.**

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350 Hanley Industrial Court, St. Louis, MO 63144-1599
314/644-1000 314/644-4617 (fax) www.carboline.com

An RPM Company

April 2007 replaces February 2007

product data **carboline**®

Carbozinc® 11

Includes Carbozinc 11 FG

Selection & Specification Data

Generic Type Solvent Based Inorganic Zinc

Description Time-tested corrosion resistant primer that protects steel galvanically in the harshest environments. For over four decades, Carbozinc 11 (CZ 11) has been the industry standard for high-performance inorganic zinc protection on steel structures worldwide.

Features

- CZ 11 and CZ 11 FG meet Class B slip co-efficient and creep testing criteria for use on faying surfaces
- Rapid cure. Dry to handle in 45 minutes at 60°F (16°C) and 50% relative humidity.
- Low temperature cure down to 0°F (-18°C).
- High zinc loading.
- Meets FDA requirements in gray color.
- Available in ASTM D520, Type II zinc version.
- Very good resistance to salting.
- May be applied with standard airless or conventional spray equipment.
- VOC compliant in certain areas

CZ 11 FG

- Lower zinc loading for economics.
- VOC compliant for shop/fabricator use only.

Color Green (0300); Gray (0700)

Finish Flat

Primers Self Priming

Topcoats Not required for certain exposures. Can be topcoated with Epoxies, Polyurethanes, Acrylics, High-Heat Silicones and others as recommended by your Carboline sales representative. Under certain conditions, a mist coat is required to minimize topcoat bubbling.

Dry Film Thickness 2.0-3.0 mils (50-75 microns). Dry film thickness in excess of 6.0 mils (150 microns) per coat is not recommended.

		CZ 11	CZ 11 FG
Solids Content	By Weight:	79% ± 2%	74% ± 2%
Zinc Content in dry film	By Weight:	85% ± 2%	79% ± 2%

Theoretical Coverage Rate
CZ 11: 1000 mil ft² (22.8 m²/l at 25 microns)
333 ft² at 3.0 mils (8.2 m²/l at 75 microns)
CZ 11 FG: 850 mil ft² (19.4 m²/l at 25 microns)
283 ft² at 3.0 mils (7.0 m²/l at 75 microns)
Allow for loss in mixing and application

VOC Values Carbozinc 11
EPA Method 24: 4.0 lbs./gal (479 g/l)
Thinned:
7 oz/gal w/ #21: 4.1 lbs./gal (492 g/l)
5 oz/gal w/ #26: 4.1 lbs./gal (492 g/l)
5 oz/gal w/ #33: 4.1 lbs./gal (492 g/l)
These are nominal values.

VOC Values Carbozinc 11 FG
EPA Method 24: 4.3 lbs./gal (515 g/l)
Thinned: For use in fabrication shops only to remain in VOC compliance in accordance with EPA Standards.
7 oz/gal w/ #21: 4.5 lbs./gal (539 g/l)
5 oz/gal w/ #26: 4.5 lbs./gal (539 g/l)
5 oz/gal w/ #33: 4.5 lbs./gal (539 g/l)
These are nominal values.

Dry Temp. Resistance
Untopcoated:
Continuous: 750°F (399°C)
Non-Continuous: 800°F (427°C)
With recommended silicone topcoats:
Continuous: 1000°F (538°C)
Non-Continuous: 1200°F (649°C)

Substrates & Surface Preparation

General Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.

Steel Non-Immersion: SSPC-SP6 and obtain a 1.0-3.0 mil (25-75 micron) angular blast profile.

Performance Data

CZ 11

Test Method	System	Results	Report #
ASTM A-325 Slip Co-efficient	Blasted steel 1 ct. CZ 11	0.668; meets requirements for Class B rating	02722
ASTM B117 Salt Spray	1 ct. CZ 11 at 2 mils dry film thickness over blasted steel	No rusting or blistering, cracking or delamination after 43000 hrs. Moderate salting of the surface only.	SR 408
ASTM D3363 Pencil Hardness	1 ct. CZ 11	Pencil Hardness "2H"	03278
AASHTO M300 Bullet Hole Immersion Paragraph 4.6.9	1 ct. CZ 11 over Abrasive blasted steel	No blistering or rusting of coating or rusting of bare steel area after 650 hrs. Immersion in 5% sodium chloride solution; 1.5" round bare area in coating.	02514

Test reports and additional data available upon written request.

Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modification to these guidelines to achieve the desired results.

General Guidelines:

Spray Application (General) The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco. Keep material under mild agitation during application. If spraying stops for more than 10 minutes, recirculate the material remaining in the spray line. Do not leave mixed primer in the hoses during work stoppages.

Conventional Spray Agitated pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, with a maximum length of 50', .070" I.D. fluid tip and appropriate air cap.

Airless Spray
Pump Ratio: 30:1 (min.)
GPM Output: 3.0 (min.)
Material Hose: 3/8" I.D. (min.)
Tip Size: .019-.023"
Output PSI: 1500-2000
Filter Size: 60 mesh
Teflon packings are recommended and available from the pump manufacturer.

Brush For touch-up of areas less than one square foot only. Use medium bristle brush and avoid rebrushing.

Roller Not recommended

October 2006 replaces September 2006 0250

Mixing & Thinning

Mixing — Power mix base, then combine and power mix as follows. Pour zinc filler very slowly into premixed base with continuous agitation. Mix until free of lumps. Pour mixture through a 30 mesh screen. DO NOT MIX PARTIAL KITS.
Tip: Sifting zinc through a window screen will aid in the mixing process by breaking up or catching dry zinc lumps.

Ratio

	CZ 11 1 Gal Kit	CZ 11 5 Gallon Kit	CZ 11 FG 4.6 Gallon Kit
Part A:	.75 gal.	3.75 gallons	3.75 gallons
Zinc Filler:	14.6 lbs.	73 lbs.	50 lbs.

Thinning — May be thinned up to 5 oz/gal (4%) with #26 for ambient and warm surfaces. For extremely warm or windy conditions, may be thinned up to 5 oz/gal (4%) with #33. In cool weather (below 40°F (4°C)), thin up to 7 oz/gal (6%) with #21. Use of thinners other than those supplied or recommended by Carboline may adversely affect product performance and void product warranty, whether expressed or implied.

Pot Life — 8 Hours at 75°F (24°C) and less at higher temperatures. Pot life ends when coating becomes too viscous to use.

Cleanup & Safety

Cleanup — Use Thinner #21 or Isopropyl Alcohol. In case of spillage, absorb and dispose of in accordance with local applicable regulations.

Safety — Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.

Ventilation — When used as a tank lining or in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. In addition to ensuring proper ventilation, appropriate respirators must be used by all application personnel.

Caution — This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

Application Conditions

Condition	Material	Surface	Ambient	Humidity
Normal	40°-95°F (4°-35°C)	40°-110°F (4°-43°C)	40°-95°F (4°-35°C)	40-90%
Minimum	0°F (-18°C)	0°F (-18°C)	0°F (-18°C)	30%
Maximum	130°F (54°C)	200°F (93°C)	130°F (54°C)	95%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

Curing Schedule

Surface Temp. & 50% Relative Humidity	Dry to Handle	Dry to Topcoat/Recoat
0°F (-18°C)	4 Hours	7 Days
40°F (4°C)	1 Hour	48 Hours
60°F (16°C)	¾ Hour	24 Hours
80°F (27°C)	¾ Hour	18 Hours
100°F (38°C)	¼ Hour	16 Hours

These times are based on a 3.0-4.0 mil (75-100 micron) dry film thickness. Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Humidity levels below 50% will require longer cure times. **Notes**: Any salting that appears on the zinc surface as a result of prolonged weathering exposure must be removed prior to the application of additional coatings. Also, loose zinc must be removed from the cured film by rubbing with fiberglass screen wire if: 1) The Carbozinc 11 is to be used without a topcoat in immersion service and "zinc pick up" could be detrimental, or 2) When "dry spray/overspray" is evident on the cured film and a topcoat will be applied. For **accelerated curing or where the relative humidity is below 40%**, allow an initial 2-hour ambient cure. Follow 2 hour cure with water misting or steam to keep the coated surface wet for a minimum of 8 hours and until the coated surface achieves a "2H" pencil hardness per ASTM D3363.

Packaging, Handling & Storage

CZ 11 Shipping Weight (Approximate) — 1 Gallon Kit: 23 lbs (10 kg); 5 Gallon Kit: 113 lbs (51 kg)

CZ 11 FG Shipping Weight (Approximate) — 4.6 Gallon Kit: 104 lbs. (47 kg)

Flash Point (Setaflash) — Part A: 55°F (13°C)
Zinc Filler: NA

Storage (General) — Store Indoors.

Storage Temperature & Humidity — 40° -100°F (4-38°C).
0-90% Relative Humidity

Shelf Life: 11 & 11FG — Part A: 12 months at 75°F (24°C)
Part B: 24 months at 75°F (24°C)

***Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.**

carboline.

350 Hanley Industrial Court, St. Louis, MO 63144-1599
314/644-1000 314/644-4617 (fax) www.carboline.com

An RPM Company

October 2006 replaces September 2006

product data **carboline.**

Thermaline® 450
Novolac

Selection & Specification Data

Generic Type Amine-Cured Novolac Epoxy

Description Highly cross-linked, glass flake-filled polymer that offers exceptional barrier protection and resistance to wet/dry cycling at elevated temperatures. Suitable for insulated and non-insulated pipes, stacks and equipment operating up to 450°F (232°C). This coating provides excellent resistance to corrosion, abrasion and permeation, and its novolac-modification resists severe chemical attack.

Features
- Temperature resistance up to 450°F (232°C)
- High-build single-coat capabilities
- Excellent resistance to thermal shock
- Superior abrasion and chemical resistance through internal reinforcement
- Ambient-temperature cure
- VOC compliant to current AIM regulations

Color Red (0500); Gray (5742)

Finish Eggshell

Primers Self-priming. May be applied over epoxies and phenolics.

Topcoats Epoxies, Polyurethanes

Dry Film Thickness 8.0-10.0 mils (200-250 microns)
Do not exceed 15 mils (375 microns) per coat.

Solids Content By Volume: 70% ± 2%

Theoretical Coverage Rate 1117 mil ft² (27.9 m²/l at 25 microns)
Allow for loss in mixing and application

VOC Values
As supplied: 2.08 lbs/gal (250 g/l)
Thinned:
13 oz/gal w/ #213: 2.58 lbs/gal (308 g/l)
13 oz/gal w/#2 2.54 lbs/gal (305 g/l)
These are nominal values.

Dry Temp. Resistance
Continuous: 425°F (218°C)
Non-Continuous: 450°F (232°C)
Discoloration and loss of gloss may be observed above 200°F (93°C).

Limitations Epoxies lose gloss, discolor and eventually chalk in sunlight exposure.

Substrates & Surface Preparation

General Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.

Steel
Non-Insulated: SSPC-SP6
Insulated: SSPC-SP10
Surface Profile: 2.0-3.0 mils (50-75 microns)

Performance Data

Test Method	System	Results	Report #
ASTM D3359 Adhesion	Blasted Steel 2 cts. 450	4A	08460
ASTM D4060 Abrasion	Blasted Steel 2 cts. 450	171 mg loss after 1000 cycles; CS17 wheel, 1000 gram load	02910
ASTM D2794 Impact	Blasted Steel 1 ct. 450	.375 in. from damaged area. 100-in./lbs	02675
Heat Cycling Test	Blasted Steel 1 ct. 450	No cracking, blistering or delamination of film after 425°F for 1 hr/ambient/ -10°F for 24 hrs/ambient/ 425°F for 24 hrs/ambient/ -10°F for 24 hrs/ambient/ 425°F for 200 hr/ambient	SR342
Modified NACE Std. TM-01-74B Immersion	Blasted Steel 2 cts. 450	No effect to coating film except discoloration after 6 month exposure, Deionized water	02551
Chemical Resistance	Blasted Steel 1 ct. 450	Resistant to fumes of commons acids, alkalis, solvents and hydrocarbon compounds. Resistant to splash and spillage of alkalis, solvents and hydrocarbons. Acid contact may cause discoloration of coating.	SR 359 02735 03133 02794

Test reports and additional data available upon written request.

January 2009 replaces June 2006 1039

Thermaline® 450 Novolac

Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modifications to these guidelines to achieve the desired results.

General Guidelines:

Spray Application (General) The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.

Conventional Spray Pressure pot equipped with dual regulators, ½" I.D. minimum material hose, .110" I.D. fluid tip and appropriate air cap.

Airless Spray

Pump Ratio: 45:1 (min.)*
GPM Output: 3.0 (min.)
Material Hose: ½" I.D. (min.)
Tip Size: .035-.041"
Output PSI: 2200-2500

*Teflon packings are recommended and available from the pump manufacturer.

Brush For striping of welds and touch-up of small areas only. Use a medium natural bristle brush and avoid rebrushing.

Roller Not recommended.

Mixing & Thinning

Mixing Power mix separately, then combine and power mix. DO NOT MIX PARTIAL KITS.

Ratio 4:1 Ratio (A to B)

Thinning May be thinned up to 13 oz/gal (10%) with Thinner #213. For application on horizontal surfaces, may be thinned up to 13 oz/gal (10)% with Thinner #2. Agitate Thinner #213 before use. Thinner #213 will have a thick viscous appearance which is normal. Use of thinners other than those supplied by Carboline may adversely affect product performance and void product warranty, whether expressed or implied.

Pot Life 3 Hours at 75°F (24°C). Pot life ends when coating loses body and begins to sag. Pot life times will be less at higher temperatures.

Cleanup & Safety

Cleanup Use Thinner #2 or Acetone. In case of spillage, absorb and dispose of in accordance with local applicable regulations.

Safety Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.

Ventilation When used in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. User should test and monitor exposure levels to insure all personnel are below guidelines. If not sure or if not able to monitor levels, use MSHA/NIOSH approved supplied air respirator.

Caution This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

Application Conditions

Condition	Material	Surface	Ambient	Humidity
Normal	65°-85°F (18°-29°C)	65°-85°F (18°-29°C)	65°-85°F (18°-29°C)	30-60%
Minimum	55°F (13°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	110°F (43°C)	100°F (38°C)	85%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

Curing Schedule

Surface Temp. & 50% Relative Humidity	Dry to Handle	Dry to Topcoat w/ Other Finishes	Final Cure
50°F (10°C)	18 Hours	48 Hours	21 Days
60°F (16°C)	12 Hours	32 Hours	14 Days
75°F (24°C)	6 Hours	16 Hours	7 Days
90°F (32°C)	3 Hours	8 Hours	4 Days

These times are based on a 10.0 mil (250 micron) dry film thickness. Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Excessive humidity or condensation on the surface during curing can interfere with the cure, can cause discoloration and may result in a surface haze. Any haze or blush <u>must</u> be removed by water washing before recoating. During high humidity conditions, it is recommended that the application be done while temperatures are increasing. If the final cure time is exceeded, the surface must be abraded by sweep blasting prior to the application of additional coats.

Packaging, Handling & Storage

Shipping Weight (Approximate) 1 Gallon Kit: 12 lbs (6 kg); 5 Gallon Kit: 58 lbs (26 kg)

Flash Point (Setaflash) Part A: 53°F (12°C); Part B: >200°F (93°C)

Storage (General) Store Indoors.

Storage Temperature & Humidity 40° - 110°F (4°-43°C); 0-90% Relative Humidity

Shelf Life Part A & B: Min. 36 months at 75°F (24°C)

***Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.**

carboline

350 Hanley Industrial Court, St. Louis, MO 63144-1599
314/644-1000 314/644-4617 (fax) www.carboline.com

An RPM Company

January 2009 replaces June 2006

Industrial & Marine Coatings

SHERWIN WILLIAMS.

ZINC CLAD® II PLUS

INORGANIC ZINC-RICH COATING

PART A	B69VZ12	BASE
PART B	B69VZ13	ACCELERATOR
PART B	B69VZ15	ACCELERATOR
PART F	B69D11	ZINC DUST

PRODUCT INFORMATION

Revised 12/05

PRODUCT DESCRIPTION

ZINC CLAD II PLUS is a solvent-based, three component, inorganic ethyl silicate, zinc rich coating. This is fast drying, high solids, low VOC coating with 82%, by weight, of zinc dust in the dry film.

- Coating self-heals to resume protection if damaged
- Provides cathodic/sacrificial protection by the same mechanism as galvanizing
- Forms an inorganic barrier to moisture and solvents
- Meets Class B requirements for Slip Coefficient and Creep Resistance, 0.67
- Meets AASHTO M-300 specification

RECOMMENDED USES

For use over prepared blasted steel and galvanized steel in areas such as:

- Bridges
- Refineries
- Shop or field application
- Drilling rigs
- As a one-coat maintenance coating or as a permanent primer for severe corrosive environments (pH range 5-9)
- Ideal for application at low temperatures or service at high temperatures and/or humidity conditions
- Fresh and demineralized water immersion service (non-potable)
- Compliance with Class B Slip Coefficient rating when used alone or as part of a system with Steel Spec Epoxy Primer as a topcoat

PRODUCT CHARACTERISTICS

Finish: Flat

Color: Gray-Green

Volume Solid: 76% ± 2%, mixed

Weight Solid: 90% ± 2%, mixed

VOC (EPA Method 24): Unreduced: <320 g/L; 2.67 lb/gal (mixed)
Reduced 4%: <340 g/L; 2.8 lb/gal

Zinc Content in Dry Film: 82% by weight

Mix Ratio: 3 components, premeasured 3.66 gallons mixed

Recommended Spreading Rate per coat:

Wet mils:	3.0 - 6.0
Dry mils:	2.0 - 4.0
Coverage:	400 - 610 sq ft/gal approximate

Note: Brush application is for small areas only.
Application of coating above maximum or below minimum recommended spreading rate may adversely affect coating performance.

Drying Schedule @ 4.0 mils wet @ 50% RH:

	@40°F	@ 77°F	@100°F
To touch:	25 minutes	20 minutes	5 minutes
To handle:	1 hour	20 minutes	15 minutes
To topcoat:	7 days	24 hours	8 hours
To cure:	7 days	36 hours	24 hours
To stack	6 hours	2 hours	1 hour

Drying time is temperature, humidity, and film thickness dependent.

Pot Life: 8 hours @ 77°F
High humidity will shorten pot life

Sweat-in-time: None required, but material should be mixed for at least 5 minutes before use

Shelf Life: Part A - 12 months, unopened
Part B - 24 months, unopened
Part F - 24 months, unopened
Store indoors at 40°F to 100°F

Flash Point (mixed): 55°F

Reducer/Clean up:
Above 70°F: R2KT4, 150 Flash Naphtha
Below 70°F: R2K4, Xylene

PERFORMANCE CHARACTERISTICS

System Tested: (unless otherwise indicated)
Substrate: Steel
Surface Preparation: SSPC-SP10
1 ct. Zinc Clad II Plus @ 3.0 mils dft

Adhesion:
Method: ASTM D4541
Result: 689 psi

Direct Impact Resistance:
Method: ASTM D2794-92
Result: 60 in lbs.

Dry Heat Resistance:
Method: ASTM D2485
Result: 750°F*

Flexibility:
Method: ASTM D522, 180° bend, 1" mandrel
Result: Passes

Pencil Hardness:
Method: ASTM D3363
Result: 3H

Salt Fog Resistance:
Method: ASTM B117, 7000 hours
Result: Rating 9 per ASTM D714 for blistering
Rating 9 per ASTM D610 for rusting

Slip Coefficient (zinc only):
Method: AISC Specification for Structural Joints Using ASTM A325 or ASTM A490 Bolts
Result: Class B, 0.67

Slip Coefficient (system listed below):
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft
1 ct. Steel Spec Epoxy Primer @ 4.0 - 6.0 mils dft
Method: AISC Specifiction for Structural Joints using ASTM A325 or ASTM A490 Bolts
Result: Passes Class B, .56

Provides performance comparable to products formulated to specifications Mil-P-38336 and Mil-P-46105.

*Acceptable for use up to 1000°F when topcoated with Kem Hi-Temp Heat-Flex II 800 Aluminum.

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Industrial & Marine Coatings

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ZINC CLAD® II PLUS

INORGANIC ZINC-RICH COATING

PART A	B69VZ12	BASE
PART B	B69VZ13	ACCELERATOR
PART B	B69VZ15	ACCELERATOR
PART F	B69D11	ZINC DUST

PRODUCT INFORMATION

RECOMMENDED SYSTEMS

Steel, Immersion:
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft

Steel, Epoxy Topcoat, Atmospheric:
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft
1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft

Steel, Polyurethane Topcoat, Atmospheric:
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft
1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft
1 ct. Acrolon 218 HS @ 3.0 - 6.0 mils dft

Steel, Polyurethane Topcoat, Atmospheric:
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft
1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft
1 ct. Hi-Solids Polyurethane @ 3.0 - 4.0 mils dft

NOTE: 1 ct. of DTM Wash Primer can be used as an intermediate coat under recommended topcoats to prevent pinholing.

Steel (Class B Compliant System):
1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft
1 ct. Steel Spec Epoxy Primer, red @ 4.0 - 6.0 mils dft

The systems listed above are representative of the product's use. Other systems may be appropriate.

SURFACE PREPARATION

Surface must be clean, dry, and in sound condition. Remove all oil, dust, grease, dirt, loose rust, and other foreign material to ensure adequate adhesion.

Refer to product Application Bulletin for detailed surface preparation information.

Minimum recommended surface preparation:
Iron & Steel:
Atmospheric: SSPC-SP6/NACE 3, 2 mil profile
Immersion: SSPC-SP10/NACE 2, 2 mil profile

TINTING

Do not tint.

APPLICATION CONDITIONS

Temperature: 20°F minimum, 100°F maximum (air, surface, and material)
At least 5°F above dew point

Relative humidity: 40% - 90% maximum
Water misting may be required at humidities below 50%

Refer to product Application Bulletin for detailed application information.

ORDERING INFORMATION

Packaging: 3.66 gallons total, mixed
Part A: 2.21 gallon kit
Part B: 0.20 gallon
Part F: 73 lbs zinc dust

Weight per gallon: 26.83 ± 0.2 lb, mixed

SAFETY PRECAUTIONS

Refer to the MSDS sheet before use.

Published technical data and instructions are subject to change without notice. Contact your Sherwin-Williams representative for additional technical data and instructions.

DISCLAIMER

The information and recommendations set forth in this Product Data Sheet are based upon tests conducted by or on behalf of The Sherwin-Williams Company. Such information and recommendations set forth herein are subject to change and pertain to the product offered at the time of publication. Consult your Sherwin-Williams representative to obtain the most recent Product Data Information and Application Bulletin.

WARRANTY

The Sherwin-Williams Company warrants our products to be free of manufacturing defects in accord with applicable Sherwin-Williams quality control procedures. Liability for products proven defective, if any, is limited to replacement of the defective product or the refund of the purchase price paid for the defective product as determined by Sherwin-Williams. NO OTHER WARRANTY OR GUARANTEE OF ANY KIND IS MADE BY SHERWIN-WILLIAMS, EXPRESSED OR IMPLIED, STATUTORY, BY OPERATION OF LAW OR OTHERWISE, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE.

Industrial & Marine Coatings

SHERWIN WILLIAMS.

6.13A

ZINC CLAD® II PLUS

INORGANIC ZINC-RICH COATING

PART A	B69VZ12	BASE
PART B	B69VZ13	ACCELERATOR
PART B	B69VZ15	ACCELERATOR
PART F	B69D11	ZINC DUST

APPLICATION BULLETIN

Revised 12/05

SURFACE PREPARATION

Zinc rich coatings require direct contact between the zinc pigment in the coating and the metal substrate for optimum performance. Surface must be dry, free from oil, dirt, dust, mill scale or other contaminants to ensure adequate adhesion.

Iron & Steel (atmospheric service):
Remove all oil and grease from surface by Solvent Cleaning per SSPC-SP1. Minimum surface preparation is Commercial Blast Cleaning per SSPC-SP6/NACE 3. For better performance, use Near White Metal Blast Cleaning per SSPC-SP10/NACE 2. Blast clean all surfaces using a sharp, angular abrasive for optimum surface profile (2 mils). Prime any bare steel the same day as it is cleaned or before flash rusting occurs.

Iron & Steel (immersion service):
Remove all oil and grease from surface by Solvent Cleaning per SSPC-SP1. Minimum surface preparation is Near White Metal Blast Cleaning per SSPC-SP10/NACE 2. Blast clean all surfaces using a sharp, angular abrasive for optimum surface profile (2 mils). Remove all weld spatter and round all sharp edges by grinding. Prime any bare steel the same day as it is cleaned or before flash rusting occurs.

Note: If blast cleaning with steel media is used, an appropriate amount of steel grit blast media may be incorporated into the work mix to render a dense, angular 1.5 - 2.0 mil surface profile. This method may result in improved adhesion and performance.

APPLICATION CONDITIONS

Temperature:	20°F minimum, 100°F maximum (air, surface, and material) At least 5°F above dew point
Relative humidity:	40% - 90% maximum Water misting may be required at humidities below 50%

APPLICATION EQUIPMENT

The following is a guide. Changes in pressures and tip sizes may be needed for proper spray characteristics. Always purge spray equipment before use with listed reducer. Any reduction must be compliant with existing VOC regulations and compatible with the existing environmental and application conditions.

Reducer/Clean up
Above 70°F .... R2KT4, 150 Flash Naphtha
Below 70°F .... R2K4, Xylene

Airless Spray
(use Teflon packings and continuous agitation)
Unit .... Graco 30:1
Pressure .... 2700 psi
Hose .... 3/8" ID
Tip .... .019" - .021"
Filter .... 30 mesh
Reduction .... As needed up to 4% by volume
For continuous operation in larger areas, use Speeflo Airless Commander Zinc Pump. Set ball checks to maximum travel for viscous material.

Conventional Spray
(continuous agitation required)
Gun .... Binks 95
Fluid Nozzle 66
Fluid Hose .... 1/2" ID, 50 ft maximum
Air Nozzle 63PB
Air Hose .... 1/2" ID, 50 ft maximum
Atomization Pressure ... 25 psi
Fluid Pressure .... 10-20 psi
Reduction .... As needed up to 4% by volume

Keep pressure pot at level of applicator to avoid blocking of fluid line due to weight of material. Blow back coating in fluid line at intermittent shutdowns, but continue agitation at pressure pot.

Brush .... For touch up in small areas only

If specific application equipment is not listed above, equivalent equipment may be substituted.

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Industrial & Marine Coatings

6.13A
ZINC CLAD® II PLUS
INORGANIC ZINC-RICH COATING

Part A	B69VZ12	Base
Part B	B69VZ13	Accelerator
Part B	B69VZ15	Accelerator
Part F	B69D11	Zinc Dust

APPLICATION BULLETIN

Application Procedures

Surface preparation must be completed as indicated.
Zinc Clad II Plus comes in premeasured containers, which when mixed provides ready-to-apply material.

Mixing Instructions:
Thoroughly agitate Binder, Part A. Using continuous air driven agitation, slowly mix all of Zinc Dust, Part F, into all of Binder Part A until mixture is completely uniform. Continue agitation and add Part B. After mixing, pour mixture through 30-mesh screen. Mixed material must be used within 8 hours. Do not mix previously mixed material with new. No "sweat-in" period is required.

If reducer solvent is used, add only after components have been thoroughly mixed.

Continuous agitation of mixture during application is required, otherwise zinc dust will quickly settle out.

Apply paint at the recommended film thickness and spreading rate as indicated below:

Recommended Spreading Rate per coat:

Wet mils:	3.0 - 6.0
Dry mils:	2.0 - 4.0
Coverage:	400 - 610 sq ft/gal approximate

Note: Brush application is for small areas only.
Application of coating above maximum or below minimum recommended spreading rate may adversely affect coating performance.

Drying Schedule @ 4.0 mils wet @ 50% RH:

	@40°F	@ 77°F	@100°F
To touch:	25 minutes	20 minutes	5 minutes
To handle:	1 hour	20 minutes	15 minutes
To topcoat:	7 days	24 hours	8 hours
To cure:	7 days	36 hours	24 hours
To stack	6 hours	2 hours	1 hour

Drying time is temperature, humidity, and film thickness dependent.

Pot Life: 8 hours @ 77°F
High humidity will shorten pot life

Sweat-in-time: None required, but material should be mixed for at least 5 minutes before use

Performance Tips

Topcoating: Note minimum cure times at normal conditions before topcoating. Longer drying periods are required if primer cannot be water mist sprayed when humidity is low. Water misting may be required at humidities below 50% to enhance cure rate.

Occasionally topcoats will pinhole or delaminate from zinc-rich coatings. This is usually due to poor ambient conditions or faulty application of topcoats. This can be minimized by:
- Provide adequate ventilation and suitable application and substrate temperature.
- If pinholing develops during topcoating, apply a mist coat of the topcoat, reduced up to 50%. Allow 10 minutes flash off and follow with a full coat.

An intermediate coat is recommended to provide uniform appearance of the topcoat.

Stripe coat all crevices, welds, and sharp angles to prevent early failure in these areas.

When using spray application, use a 50% overlap with each pass of the gun to avoid holidays, bare areas, and pinholes. If necessary, cross spray at a right angle.

Spreading rates are calculated on volume solids and do not include an application loss factor due to surface profile, roughness or porosity of the surface, skill and technique of the applicator, method of application, various surface irregularities, material lost during mixing, spillage, overthinning, climatic conditions, and excessive film build.

Excessive reduction of material can affect film build, appearance, and performance.

Do not mix previously catalyzed material with new.

Do not apply the material beyond recommended pot life.

In order to avoid blockage of spray equipment, clean equipment before use or before periods of extended downtime with Reducer R2KT4, 150 Flash Naphtha.

Keep pressure pot at level of applicator to avoid blocking of fluid line due to weight of material. Blow back coating in fluid line at intermittent shutdowns, but continue agitation at pressure pot.

Application above recommended film thickness may result in mud cracking and poor topcoat appearance.

During the early stages of drying, the coating is sensitive to rain, dew, high humidity, and moisture condensation. If possible, plan painting schedules to avoid these influences during the first 16-24 hours of curing.

Topcoats may be applied once 50 MEK double rubs are achieved. No zinc or only slight traces should be visible. Coin hardness test can also be used.

Refer to Product Information sheet for additional performance characteristics and properties.

Clean Up Instructions

Clean spills and spatters immediately with Reducer R2KT4, 150 Flash Naphtha or R2K4, Xylene. Clean hands and tools immediately after use with Reducer R2KT4, 150 Flash Naphtha or R2K4, Xylene. Follow manufacturer's safety recommendations when using any solvent.

Safety Precautions

Refer to the MSDS before use.

Published technical data and instructions are subject to change without notice. Contact your Sherwin-Williams representative for additional technical data and instructions.

Disclaimer

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Warranty

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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.

The operations described in this chapter assume that the fuel will be loaded into or unloaded from the MPC submerged in a spent fuel pool. With some modifications, the information presented herein can be used to develop site-specific procedures for loading or unloading fuel into the system within a hot cell or other remote handling facility.

Users are required to develop or modify existing programs and procedures to account for the implementation of the HI-STORM FW system. Written procedures are required to be developed or modified to account for such items as handling and storage of systems, structures and components identified as *important-to-safety*, heavy load handling, specialized instrument calibration, special nuclear material accountability, fuel handling procedures, training, equipment, and process qualifications. Users shall implement controls to ensure that all critical set points do not exceed the design limit of lifting equipment and appurtenances.

Control of the operation shall be performed in accordance with the user's Quality Assurance (QA) program to ensure critical steps are not overlooked and that the cask has been confirmed to meet all requirements of the CoC before being released for on-site storage under Part 72.

Fuel assembly selection and verification shall be performed by the user in accordance with written, approved procedures that ensure that only SNF assemblies authorized in the CoC are loaded into the MPC. Fuel handling shall be performed in accordance with written site-specific procedures.

ALARA notes and warnings in this chapter are included to alert users to radiological issues. Actions identified with these notes and warnings are of an advisory nature and shall be implemented based on a site-specific determination by radiation protection personnel.

Section 9.1 provides a technical basis for loading and unloading procedures. Section 9.2 provides the guidance for loading the HI-STORM FW system. Section 9.3 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 9.4 provides the procedure for unloading the HI-STORM FW system.

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.

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.

Table 9.1.1

OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6. Procedural guidance is given for cask handling, inspection of lifting equipment, and proper engagement.
Cask Tip-Over Prior to welding of the MPC lid	The design of the Lift Yoke prevents inadvertent disconnection during periods where it is attached.
Contamination of the MPC external shell	The annulus seal, bottom lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water.
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.
Damage to fuel assembly cladding from oxidation	Fuel assemblies are not directly exposed to air or oxygen during loading and unloading operations. Fuel will be blanketed with an inert gas when not immersed in water. Water is introduced at a slow rate to avoid thermal shocking of the system.
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum gauges will be isolated from pressurized gas and water systems when not used for vacuum. Isolation valves allow gauges to be easily replaced in service.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to and during welding or cutting activities. The space below the MPC lid shall be purged prior to and during these activities.
Excess dose from failed fuel assemblies during unloading operations	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.

Table 9.1.1

OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Excess generation of radioactive waste	The HI-STORM FW system uses process systems that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Where possible, items are installed by hand and require no tools.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in a controlled manner to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. If the forced helium dehydration process is used, the temperature of the gas exiting the demoisturizer is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations.
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.
Load Drop	Rigging diagrams and procedural guidance are provided to users for all applicable lifts. Component weights are provided to users on a site-specific basis. Heavy loads are handled in accordance with the guidance of NUREG-0612.
Over-pressurization of MPC during loading and unloading	Pressure relief devices in the water and gas processing systems limit the MPC pressure to acceptable levels.
Overstressing MPC lift lugs from side loading	Procedural guidance is provided for all heavy load handling activities on a site-specific basis.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts. Site-specific weight evaluations are provided.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.

Table 9.1.1

OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during moisture removal and helium backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.

9.2 PROCEDURE FOR LOADING THE HI-STORM FW SYSTEM FROM SNF IN THE SPENT FUEL POOL

9.2.1 Overview of Loading Operations

The HI-STORM FW system is used to load, transfer, and store spent fuel. Specific steps, required to prepare the HI-STORM FW system for fuel loading, to load the fuel, to prepare the system for storage, and to place it in storage at an ISFSI are described in this chapter. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC VW and/or HI-STORM FW may be moved between the ISFSI and the fuel loading facility using any load handling equipment designed for such applications. Users of the HI-STORM FW system are required to develop detailed written procedures to control on-site transport operations. Instructions for general lifting, handling, and placement of the HI-STORM FW overpack, MPC, and HI-TRAC VW vary by site and are provided on a site-specific basis in Holtec-approved procedures and instructions.

The broad operational steps are explained below and illustrative figures are provided at the end of this section. At the start of loading operations, an empty MPC is upended. The empty MPC is raised and inserted into the HI-TRAC VW. The annulus is filled with plant demineralized water[1] and an inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC VW to prevent spent fuel pool water from contaminating the exterior surface of the MPC when it is submerged in the pool. The MPC is filled with either spent fuel pool water or plant demineralized water (borated as required)[2]. The HI-TRAC VW top flange is outfitted with the lift blocks and the HI-TRAC VW and MPC are then raised and lowered into the spent fuel pool[3] for fuel loading using the lift yoke. Pre-selected assemblies[4] are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielded lid (the MPC lid) is installed. The lift yoke remotely engages to the HI-TRAC VW lift blocks 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.

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1 Users may substitute domestic water or radiologically clean borated water in each step where demineralized water is specified.

2 Users may also fill the MPC with water during HI-TRAC placement in the spent fuel pool.

3 Spent Fuel Pool as used in this chapter generically refers to the users designated cask loading location.

4 Damaged fuel assemblies are loaded and stored in Damaged Fuel Containers in the MPC basket.

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.

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. 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 is required for high-burn-up fuel at higher decay heat (it can be used as an alternative to vacuum drying) 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.

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.

Next, the HI-TRAC VW bottom lid is removed and the mating device drawer is opened. The MPC is transferred into HI-STORM FW. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. Next, the HI-TRAC VW is removed from the top of HI-STORM FW[5]. The MPC slings and MPC lift attachments are removed. Plugs are installed in the empty MPC lifting holes to fill the voids left by the lift attachment bolts. Next, the mating device is removed. The HI-STORM FW lid, along with the temperature elements (if used), and vent screens may be installed at anytime after the mating device is removed. The HI-STORM FW is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM FW overpack and HI-TRAC VW transfer cask may be moved using a number of methods as long as the lifting equipment requirements of this FSAR are met. Finally, the temperature elements connections are installed (if used), final dose rate measurements are taken, and any thermal testing (if required) is performed to ensure that the system is functioning within its design parameters.

9.2.2 Preparation of HI-TRAC VW and MPC

Note:
Handling of loaded equipment shall only be performed if the ambient temperature is above 0°F

1. Place HI-TRAC VW in the cask receiving area.

2. Perform a HI-TRAC VW receipt inspection and cleanliness inspection (See Table 9.2.5 for example).

3. Clear the HI-TRAC VW top for installation of the MPC.

4. Remove any road dirt. Remove any foreign objects from cavity locations.

5. If necessary, perform a radiological survey of the inside of HI-TRAC VW to verify there is no residual contamination from previous uses of the cask.

6. If necessary, configure HI-TRAC VW with the bottom lid.

5 The empty HI-TRAC VW may be removed from the mating device with its bottom lid installed or removed.

7. Perform an MPC receipt inspection and cleanliness inspection (See Table 9.2.4 for example).

8. Install the MPC inside HI-TRAC VW in accordance with site-approved rigging procedures.

9. If necessary, perform an MPC, lid, closure ring, drain line, vent, and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

Note:
Annulus filling and draining operations vary by site. Instructions for filling and draining the annulus along with the use of the annulus overpressure system are provided on a site-specific basis.

10. Fill the annulus with non-contaminated water to just below the inflatable seal seating surface.

11. Install the inflatable annulus seal around the MPC.

12. To the extent practicable, apply waterproof tape over any empty bolt holes or locations where water may create a decontamination issue.

Note:
Canister filling and draining operations vary by site. Instructions are provided on a site-specific basis.

13. Fill the MPC with water to approximately 12 inches below the top of the MPC shell. Refer to LCO 3.3.1 for boron concentration requirements.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

14. Place HI-TRAC VW in the designated cask loading area.

15. Verify spent fuel pool for boron concentration requirements in accordance with LCO 3.3.1. Testing must be completed within four hours prior to loading and every 48 hours after in accordance with the LCO. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

9.2.3 MPC Fuel Loading

Note:
When loading an MPC requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC. Perform a verification of the types, amounts, and location of non-fuel hardware using plant fuel records to ensure that only non-fuel hardware that meet the conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC.

2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern[6].

3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

4. If required, install fuel shims where necessary in the fuel cells.

9.2.4 MPC Closure

1. Install MPC lid and remove the HI-TRAC VW from the spent fuel pool as follows:

    a. Rig the MPC lid for installation in the MPC in accordance with site-approved rigging procedures.

    b. Install the drain line to the underside of the MPC lid.

    c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC for installation.

    d. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

    e. Record the time to begin the time-to-boil monitoring, if necessary.

    f. Engage the lift yoke to HI-TRAC VW.

6 Damaged fuel must be loaded into Damage Fuel Containers in the MPC basket.

ALARA Note:
Activated debris may have settled on the top face of HI-TRAC VW and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. Soluble boron concentration, when applicable, shall be monitored to prevent non-compliance with the Technical Specification LCO 3.3.1.

g. Raise the HI-TRAC VW until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Remove any activated or highly radioactive particles from the HI-TRAC VW or MPC.

h. Continue to raise the HI-TRAC VW under the direction of the plant's radiological control personnel. Continue general decontamination activities.

i. Remove HI-TRAC VW from the spent fuel pool while performing outer decontamination activities in accordance with directions from the radiological control personnel.

j. Place HI-TRAC VW in the designated cask preparation area.

Note:
If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with 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.

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.

a. Repair any weld defects in accordance with the applicable code requirements and re-perform the NDE in accordance with approved procedures.

6. Drain the MPC and terminate time-to-boil monitoring and boron sampling program, where required.

Note:
Detailed procedures for MPC drying are provided on a site-specific basis. The following summarize those procedures.

7. Dry and backfill the MPC (Vacuum Drying Method).

Note:
During drying activities, the annulus between the MPC and the HI-TRAC VW must be maintained full of water. Water lost due to evaporation or boiling must be replaced to maintain the water level.

a. Fill the annulus between the MPC and HI-TRAC VW with clean water. The water level must be within 6" of the top of the MPC.

b. Attach the vacuum drying system (VDS) to the vent and drain port RVOAs. Other equipment configurations that achieve the same results may also be used.

Caution:
Rapidly reducing the pressure in the VDS piping and MPC while the system contains significant amounts of water can lead to freezing of the water and to improper conclusions that the system is dry. To prevent freezing of water, the MPC internal pressure should be lowered in a controlled fashion. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

c. Start the VDS system and slowly reduce the MPC pressure to below 3 torr.

Note:
Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. If at any time during final closure operations the helium backfill gas is lost or oxidizing gases are introduced into the MPC, then the dryness test shall be repeated and the MPC refilled with helium in accordance with the Technical Specifications.

d. Perform the MPC drying pressure test in accordance with the Technical Specifications.

e. When the MPC is dry, in accordance with the acceptance criteria in the LCO 3.1.1, close the vent and drain port valves.

f. Backfill the MPC in accordance with LCO 3.1.1 using site-specific procedures.

g. Disconnect the VDS from the MPC.

h. Close the drain port RVOA cap and remove the drain port RVOA.

i. If used, stop the water flow through the annulus between the MPC and HI-TRAC. Drain.

j. Close the vent port RVOA and disconnect the vent port RVOA.

8. Dry and Backfill the MPC (FHD Method):

Note: Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. When using the FHD system to perform the MPC helium backfill, the FHD system shall be evacuated or purged and the system operated with high purity helium.
Note: MPC internal pressure during FHD operation must comply with Technical Specification.
Caution: MPC internal pressure during FHD operation may be less than the Technical Specification minimum backfill requirement. In the event of an FHD System failure where the MPC internal pressure is below the Technical Specification limit, the MPC internal pressure must be raised to at least 20 psig to place the MPC in an acceptable condition.

a. Attach the moisture removal system to the vent and drain port RVOAs. Other equipment configurations that achieve the same results may also be used.

b. Drain the water from the annulus.

c. Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.

d. Continue the monitoring and moisture removal until LCO 3.1.1 is met for MPC dryness.

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.

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

i. 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 and re-perform the NDE until the weld meets the required acceptance criteria.

10. 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.

b. Flush the cavity with helium to remove the air and immediately install the set screws recessed approximately ¼ inch below the top of the cover plate.

c. Plug weld the recess above each set screw 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.

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.

3. Perform a HI-STORM FW receipt inspection and cleanliness inspection in accordance with a site-approved inspection site-approved inspection checklist, if required.

Note:
MPC transfer may be performed at any location deemed appropriate by the licensee. The following steps describe the general transfer operations. The HI-STORM FW may be positioned on an air pad, roller skid or any other suitable equipment in the cask receiving area or at the ISFSI. The HI-STORM FW or HI-TRAC VW may be transferred to the ISFSI using any equipment specifically designed for such a function. The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

9.2.6 Placement of HI-STORM FW into Storage

1. Position an empty HI-STORM FW module at the designated MPC transfer location.

2. Remove any road dirt with water. Remove any foreign objects from cavity locations.

3. Transfer the HI-TRAC VW to the MPC transfer location.

4. Install the mating device on top of the HI-STORM FW.

5. Position HI-TRAC VW above HI-STORM FW.

6. Align HI-TRAC VW over HI-STORM FW and mate the components.

7. Attach the MPC to the lifting device in accordance with the site-approved rigging procedures.

8. Raise the MPC slightly to remove the weight of the MPC from the mating device.

9. Remove the bottom lid from HI-TRAC VW using the mating device.

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the HI-STORM FW annulus when HI-TRAC VW is removed due to radiation streaming. The mating device may be used to supplement shielding during removal of the MPC lift rigging.

10. Lower the MPC into HI-STORM FW.

11. Disconnect the MPC lifting slings from the lifting device.

Note:
It may be necessary, due to site-specific circumstances, to move HI-STORM FW from under the empty HI-TRAC VW to install the HI-STORM FW lid, while inside the Part 50 facility. In these cases, users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

12. Remove HI-TRAC VW from on top of HI-STORM FW with or without the HI-TRAC bottom lid.

13. Remove the MPC lift rigging and install plugs in the empty MPC bolt holes.

14. Place HI-STORM FW in storage as follows:

Note:
Closing the mating device drawer while the MPC is in the HI-STORM will block air flow. The mating device drawer shall remain open, to the extent possible, such that the open air path is at least as large as the HI-STORM Lid vent openings until the mating device is to be removed from the HI-STORM. When the mating device drawer is closed for mating device removal, the process shall be completed in an expeditious manner.

   a. Remove the mating device.

   b. Inspect the HI-STORM FW lid studs and nuts or lid closure bolts for general condition. Replace worn or damaged components with new ones.

Note:
Unless the lift has redundant drop protection features (or equivalent safety factor) for the HI-STORM FW lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM FW lid drop.

c. Install the HI-STORM FW lid and the lid studs and nuts or lid closure bolts.

d. Remove the HI-STORM FW lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location.

Warning:
HI-STORM FW dose rates are measured to ensure they are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

e. Perform the HI-STORM FW 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 10CFR72.104(a).

f. Secure HI-STORM FW to the transporter device as necessary.

Note:
The site-specific transport route conditions must satisfy the requirements of the Technical Specification.

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

h. Transfer the HI-STORM FW to its designated storage location at the appropriate pitch.

i. Attach the HI-STORM FW temperature elements (if used) and screens.

15. If required per CoC Condition #8 the user must perform the following annular air flow thermal test or cite a test report that was performed and prepared by another user.

a. The annular air flow thermal test shall be conducted at least 7 days after the HI-STORM is loaded in order for the overpack to establish thermal equilibrium.

b. The user or other qualified engineer shall calculate and record the actual heat load of the fuel stored in the HI-STORM.

c. To minimize the effects on the annular air flow, the test shall be performed when the weather is relatively dry and calm.

d. The ambient air temperature at the cask shall be recorded.

e. The test data shall be collected for the annular flow in the approximate center of the outlet vent as follows:

1. The outlet vent screen and gamma shield shall be removed from one outlet vent, if necessary.
2. A hot wire anemometer or similar flow measuring instrument shall be inserted into the annular space between the MPC and HI-STORM inner shell.
3. The flow measuring instrument shall be at positioned at least 6" below the top of the MPC and shall not significantly block the air flow.
4. The instrument shall not be placed too close to the MPC or HI-STORM shells to avoid edge effects on the flow.
5. The outlet gamma shield and vent screen shall be re-installed if removed.
6. Measurements of the air flow shall be taken and recorded for a minimum of three places radially across the annular gap.
7. The outlet vent screen and gamma shield shall be removed from the outlet vent, if necessary, and the flow measuring instrument removed.
8. The outlet gamma shield and vent screen shall be re-installed if removed.

f. Air flow in each of the three remaining outlet vents shall be measured and recorded in accordance with step 23.e above.

g. All test data shall be transmitted to the general license holder for evaluation and validation of the thermal model.

h. Users shall forward test and analysis results to the NRC in accordance with 10 CFR 72.4.

Table 9.2.1

HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
Air Pads/Rollers	Not Important To Safety	Used for HI-STORM FW or HI-TRAC VW cask positioning. May be used in conjunction with the cask transporter or other HI-STORM FW or HI-TRAC VW lifting device.
Annulus Overpressure System	Not Important To Safety	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Automated Welding System	Not Important To Safety	Used for remote field welding of the MPC.
Cask Transporter	Not Important to Safety unless used for MPC transfers	Used for handling of the HI-STORM FW overpack and/or the HI-TRAC VW Transfer Cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function. May also be used for MPC transfers if appropriately configured.
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	Used for rigging components such as the HI-TRAC VW top lid, bottom lid, MPC lid, AWS, and HI-STORM FW Lid and the empty MPC.
Helium Backfill System	Not Important To Safety	Used for controlled insertion of helium into the MPC for pressure testing, blowdown and placement into storage.
HI-STORM FW Lifting Devices	Determined site-specifically based on type, location, and height of lift being performed. Lifting devices shall be provided in accordance with ANSI N14.6.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM FW for cask handling.
HI-TRAC VW Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC VW handling shall be provided in accordance with ANSI N14.6.	Used for connecting the crane (or other primary lifting device) to the HI-TRAC VW for cask handling. Does not include the crane hook (or other primary lifting device). May include one or more extensions to prevent immersion of the crane hook into the spent fuel pool water.
HI-TRAC VW transfer frame	Not Important To Safety	A steel frame used to support HI-TRAC VW during delivery, on-site movement and upending/downending operations.
Inflatable Annulus Seal	Not Important To Safety	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.

Table 9.2.1

HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
MPC Lift Attachments	Important To Safety – Category A. MPC Lift Attachments shall be provided in accordance with ANSI N14.6.	MPC lift attachments consist of the strongback and attachment hardware. The MPC lift attachments are used to support the MPC during MPC transfer from HI-TRAC VW into HI-STORM FW and vice versa. The ITS classification of the lifting device attached to the attachments may be lower than the attachment itself, as determined site-specifically. Lift Attachments may take different forms based on site specific needs and may include remote disconnect features.
Pressure Test System	Not Important to Safety	Used to pressure test the MPC lid-to-shell weld.
HI-TRAC Lift Block	Important-To-Safety Category A. Lift Blocks shall be provided in accordance with ANSI N14.6.	Used to attach the HI-TRAC to the lifting yoke.
Mating Device	Important-To-Safety – Category B	Used to mate HI-TRAC VW to HI-STORM FW during transfer operations. Used to shield operators during MPC transfer operations. Includes sliding drawer for use in removing HI-TRAC VW bottom lid.
MPC Lifting Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	Used to secure the MPC to the overhead lifting device during HI-TRAC VW bottom lid removal and MPC transfer operations. Attaches between the MPC lift attachments and the lift yoke or overhead lifting device.
MPC Upending Device	Not Important to Safety	Used to evenly support the MPC during handling and upending operations and help control the upending process.
MSLD (Helium Leakage Detector)	Not Important to Safety	Used for helium leakage testing of the MPC closure welds.
Vacuum Drying System	Not Important To Safety	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Not Important To Safety	Used for removal of residual moisture from the MPC following water draining.
Vent and Drain RVOAs	Not Important To Safety	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves.
Weld Removal System	Not Important To Safety	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

Table 9.2.2

HI-STORM FW SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Instrument	Function
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors fluid flow rate during various loading and unloading operations.
Helium Mass Spectrometer Leakage Detector (MSLD)	Ensures leakage rates of welds are within acceptable limits.
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Vacuum Gages (Optional)	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

† All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 9.2.3 HI-STORM FW SYSTEM OVERPACK INSPECTION CHECKLIST
Note: This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM FW overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM FW Overpack Lid:

1. Lid studs and nuts or lid closure bolts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device/ holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.
7. Vent screens shall be inspected for proper fit and for tears and holes that would allow debris entry into the vent openings.
8. Vent openings shall be inspected for foreign material that may cause vent blockage.

HI-STORM FW Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be inspected for proper fit and for tears and holes that would allow debris entry into the vent openings.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.

Table 9.2.4 MPC INSPECTION CHECKLIST
Note: This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:

1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Vent and Drain attachments shall be inspected for availability, thread condition operability, and general condition.
4. Fuel spacers (if used) shall be inspected for availability and general condition.
5. Drain and vent port cover plates shall be inspected for availability and general condition.
6. Serial numbers shall be inspected for readability.
7. The MPC lid lift holes shall be inspected for thread condition.
8. The MPC lid, cover plates, and closure ring shall be checked for proper fit-up.

MPC Main Body:

1. All visible MPC body surfaces shall be inspected for dents, gouges, or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents, and general condition.
3. Basket panels shall be inspected for gross deformation that may inhibit fuel assembly insertion.
4. Lift lugs shall be inspected for general condition.
5. Lift lug threads shall be in inspected for thread condition
6. Verify proper MPC basket type for contents.
7. Serial numbers shall be inspected for readability.

Table 9.2.5 HI-TRAC VW TRANSFER CASK INSPECTION CHECKLIST
Note: This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC VW Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation, and potential corrective action prior to use.

<u>HI-TRAC VW Main Body:</u>

1. The painted surfaces shall be inspected for corrosion, chipped, cracked, or blistered paint.
2. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs, or any other condition that may damage the inflatable seal.
3. The nameplate shall be inspected for presence and general condition.
4. The neutron shield jacket shall be inspected for leaks.
5. Neutron shield jacket pressure relief device shall be inspected for presence and general condition.
6. The neutron shield jacket fill and neutron shield jacket drain plugs shall be inspected for presence, leaks, and general condition.
7. Bottom lid flange surface shall be clean and free of large scratches and gouges that may inhibit sealing of the lid to body.
8. The threaded anchor locations shall be inspected for thread damage, excessive wear, and general condition.

<u>HI-TRAC VW Bottom lid:</u>

1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. The painted surfaces shall be inspected for corrosion, chipped, cracked, or blistered paint.
6. Threads shall be inspected for indications of damage.

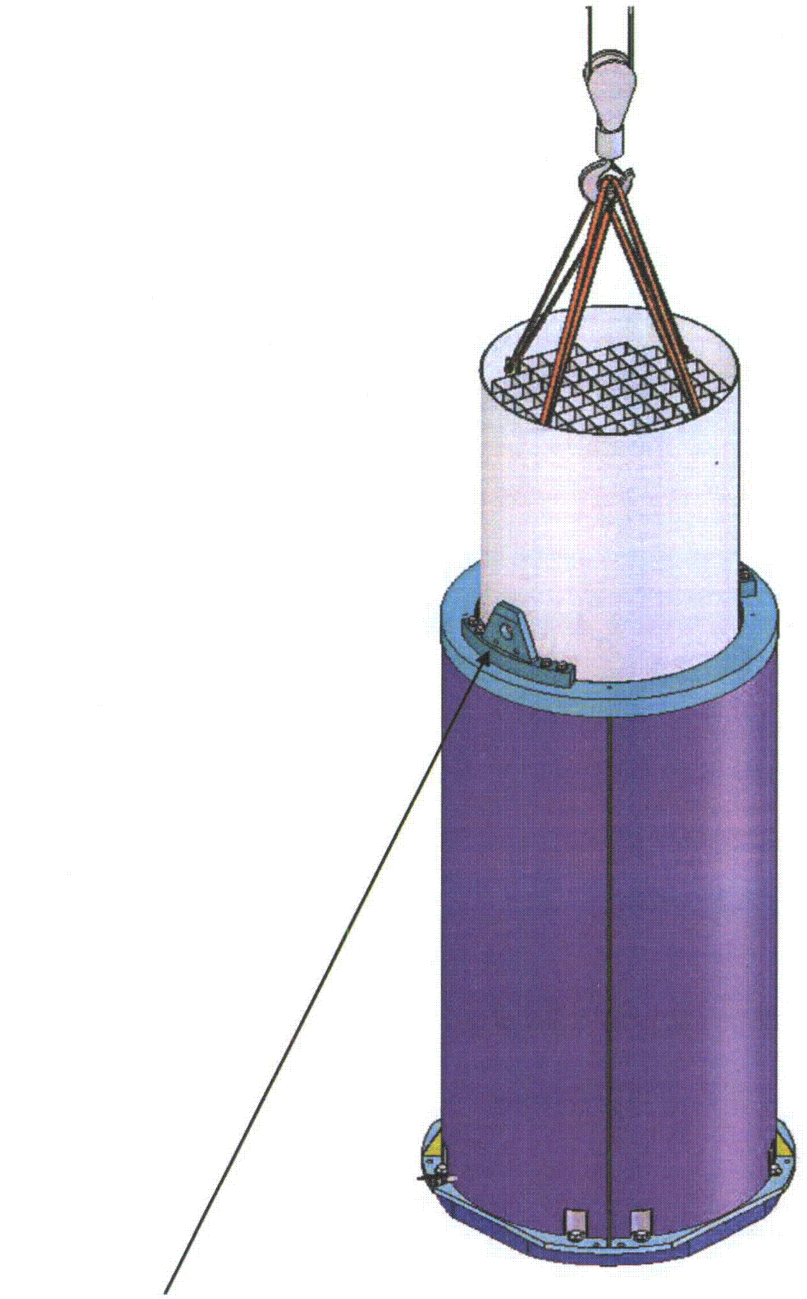

FIGURE 9.2.1: MPC INSTALLATION IN HI-TRAC

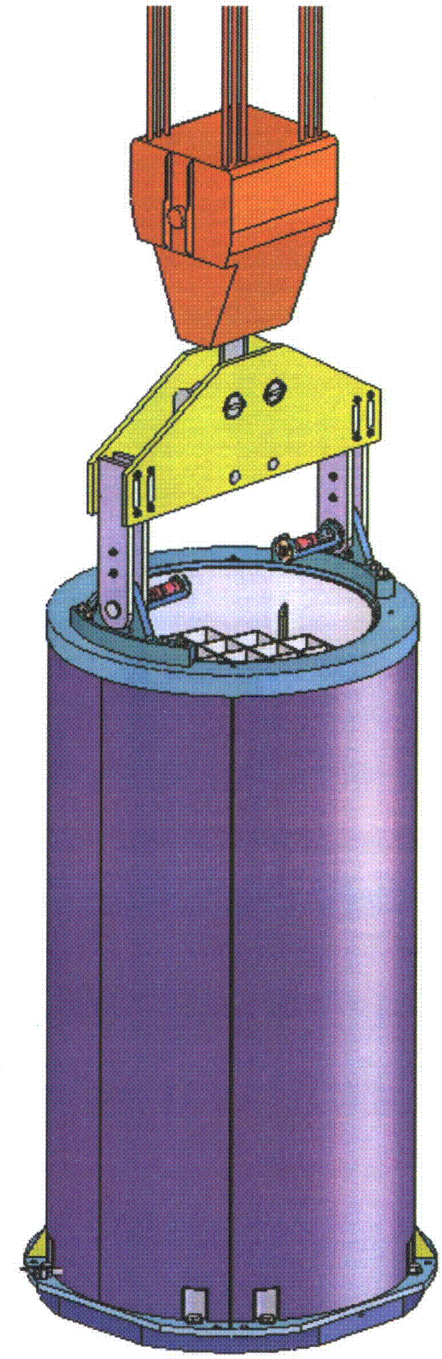

FIGURE 9.2.2: HI-TRAC LIFTING SHOWN USING A REPRESENTATIVE LIFT YOKE

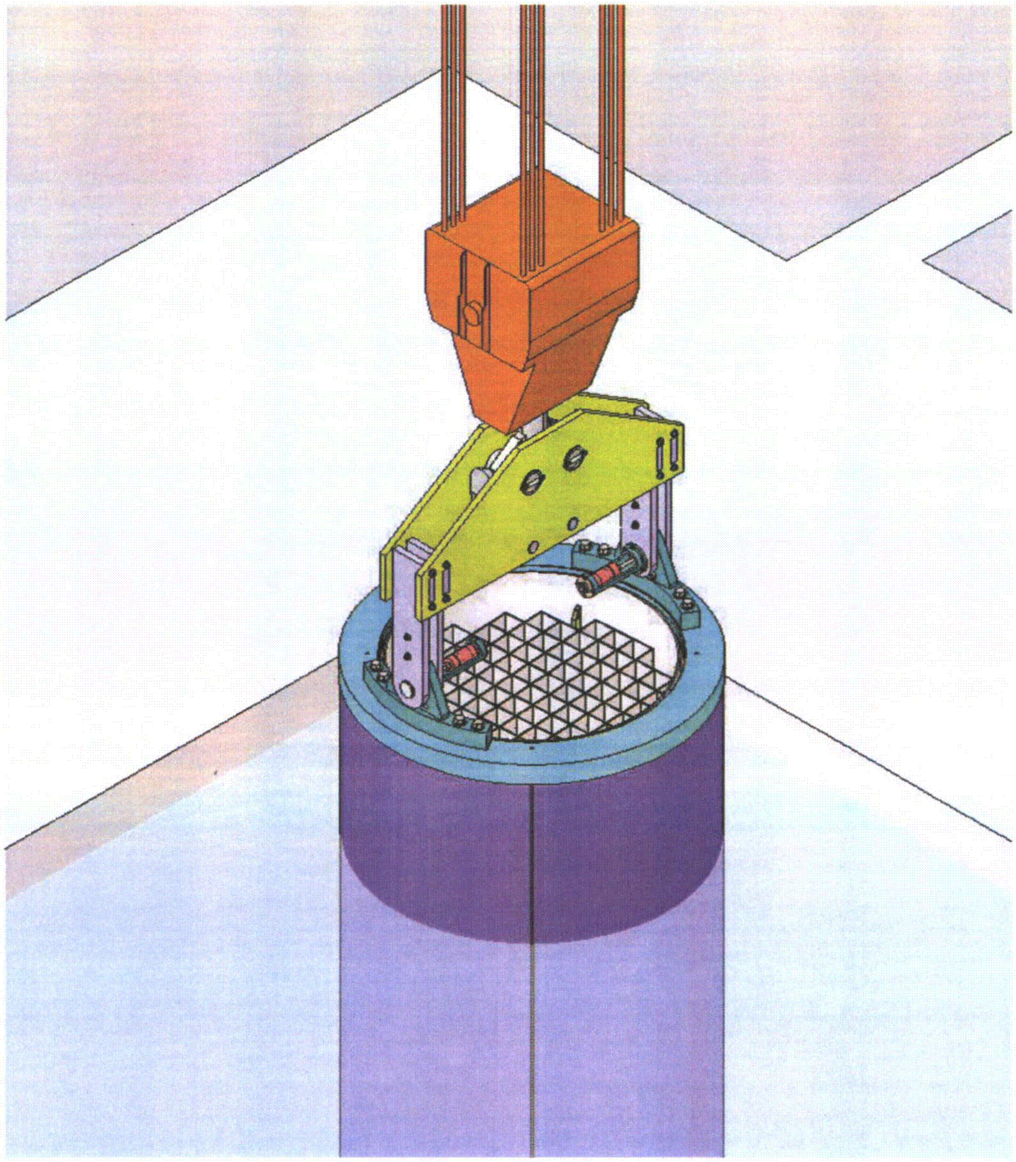

FIGURE 9.2.3: HI-TRAC PLACEMENT IN THE SPENT FUEL POOL

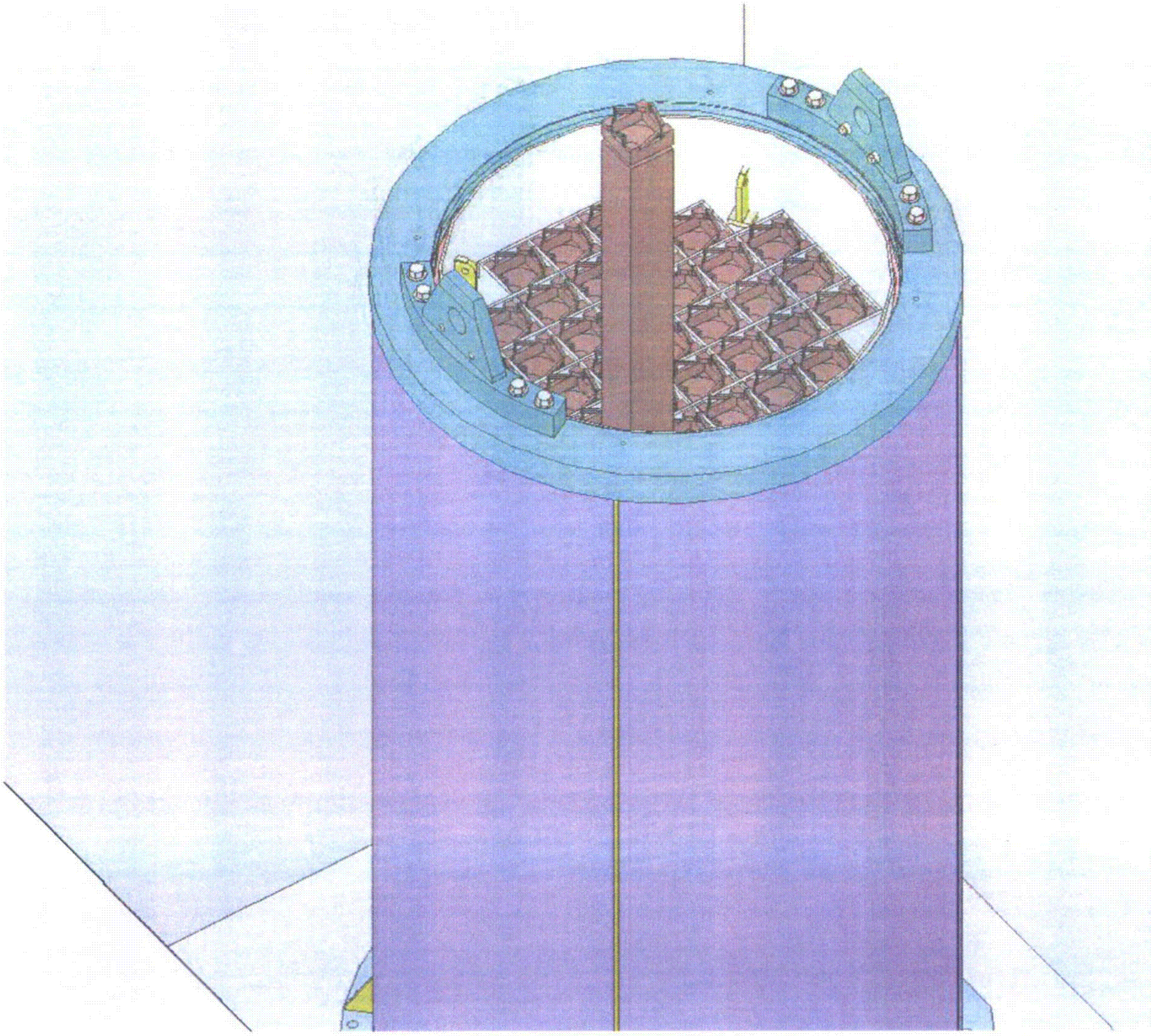

FIGURE 9.2.4: FUEL ASSEMBLY PLACEMENT IN THE MPC (CRANE NOT SHOWN)

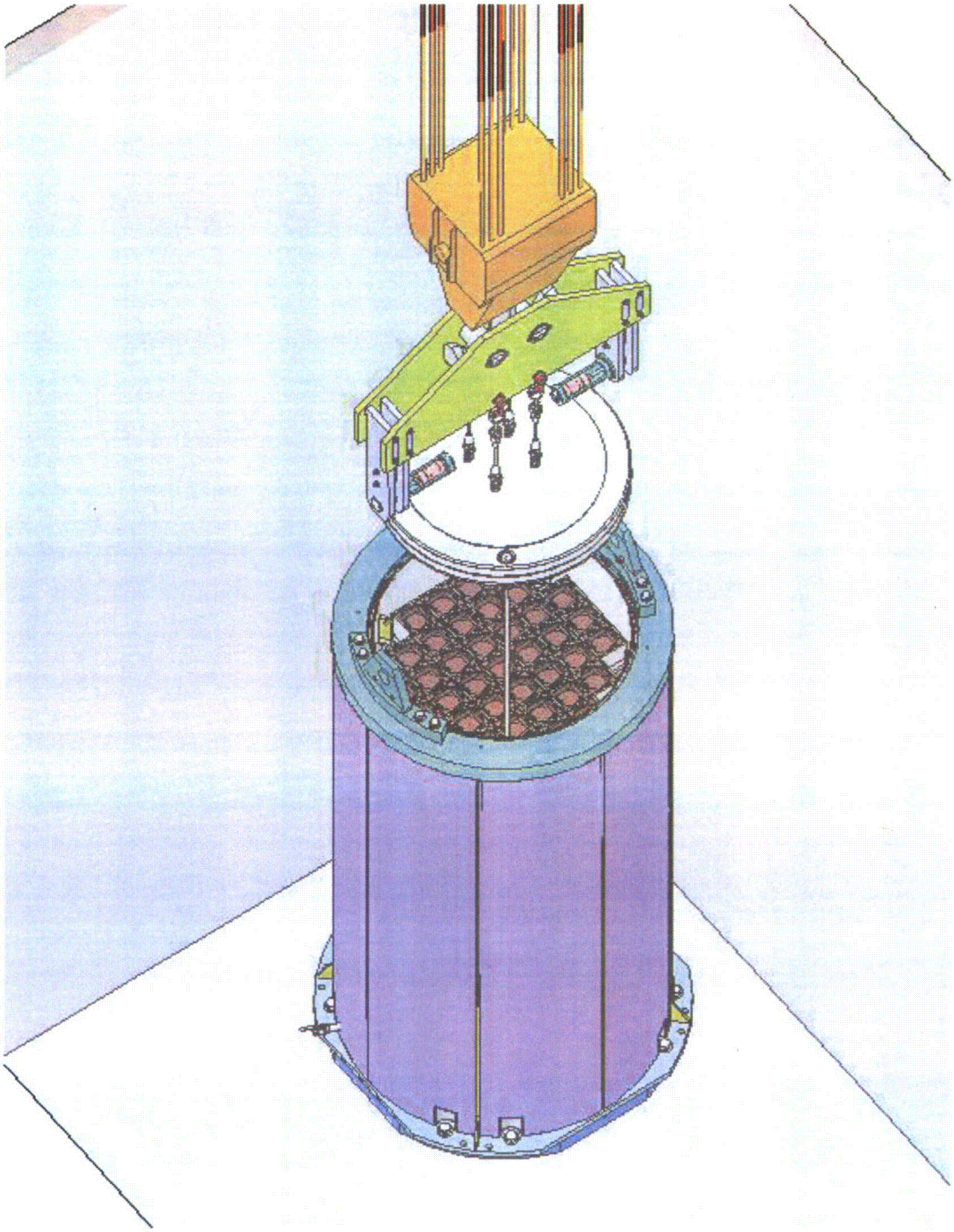

FIGURE 9.2.5: MPC LID INSTALLATION USING THE LIFT YOKE

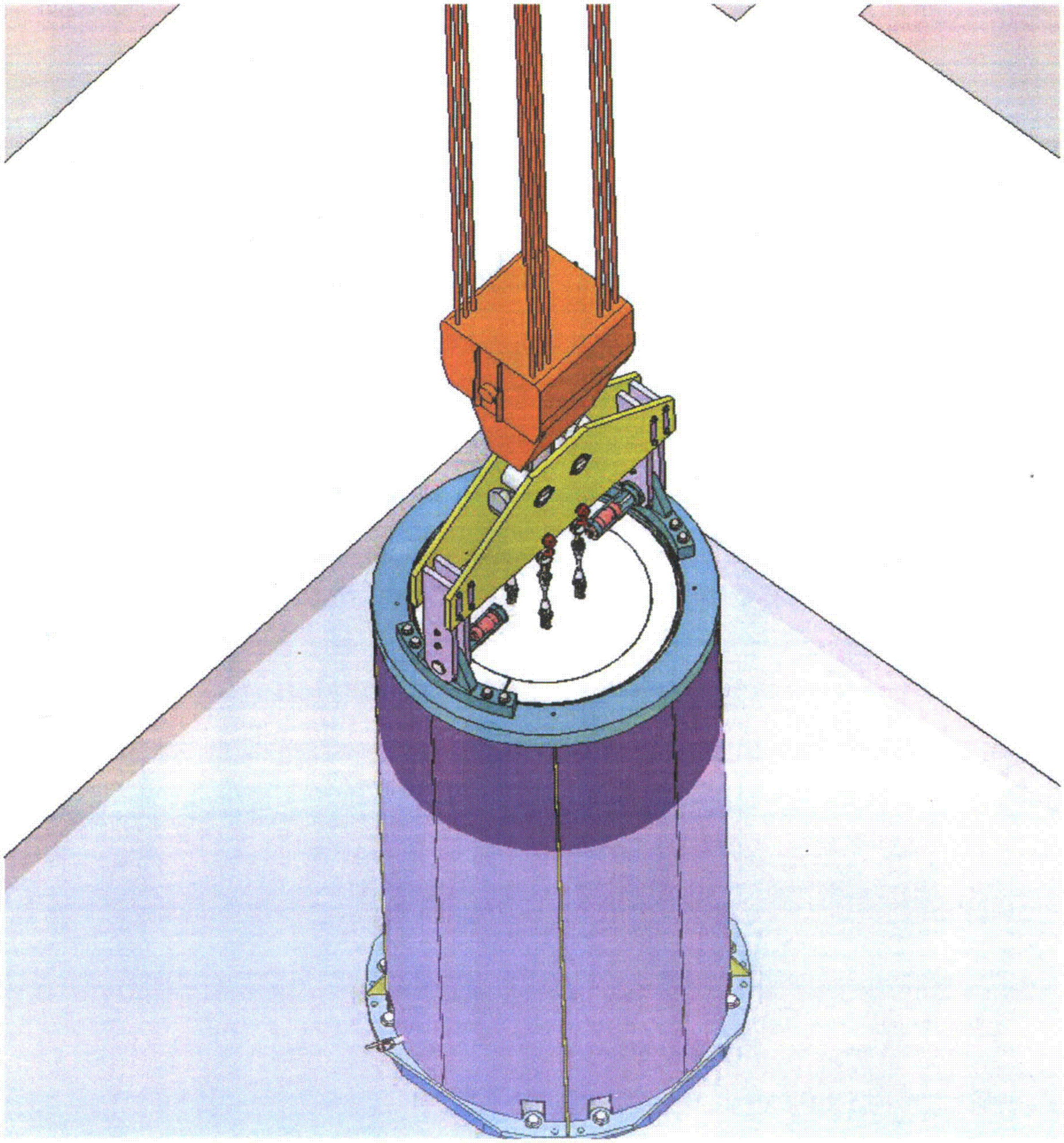

FIGURE 9.2.6: HI-TRAC REMOVAL FROM THE SPENT FUEL POOL

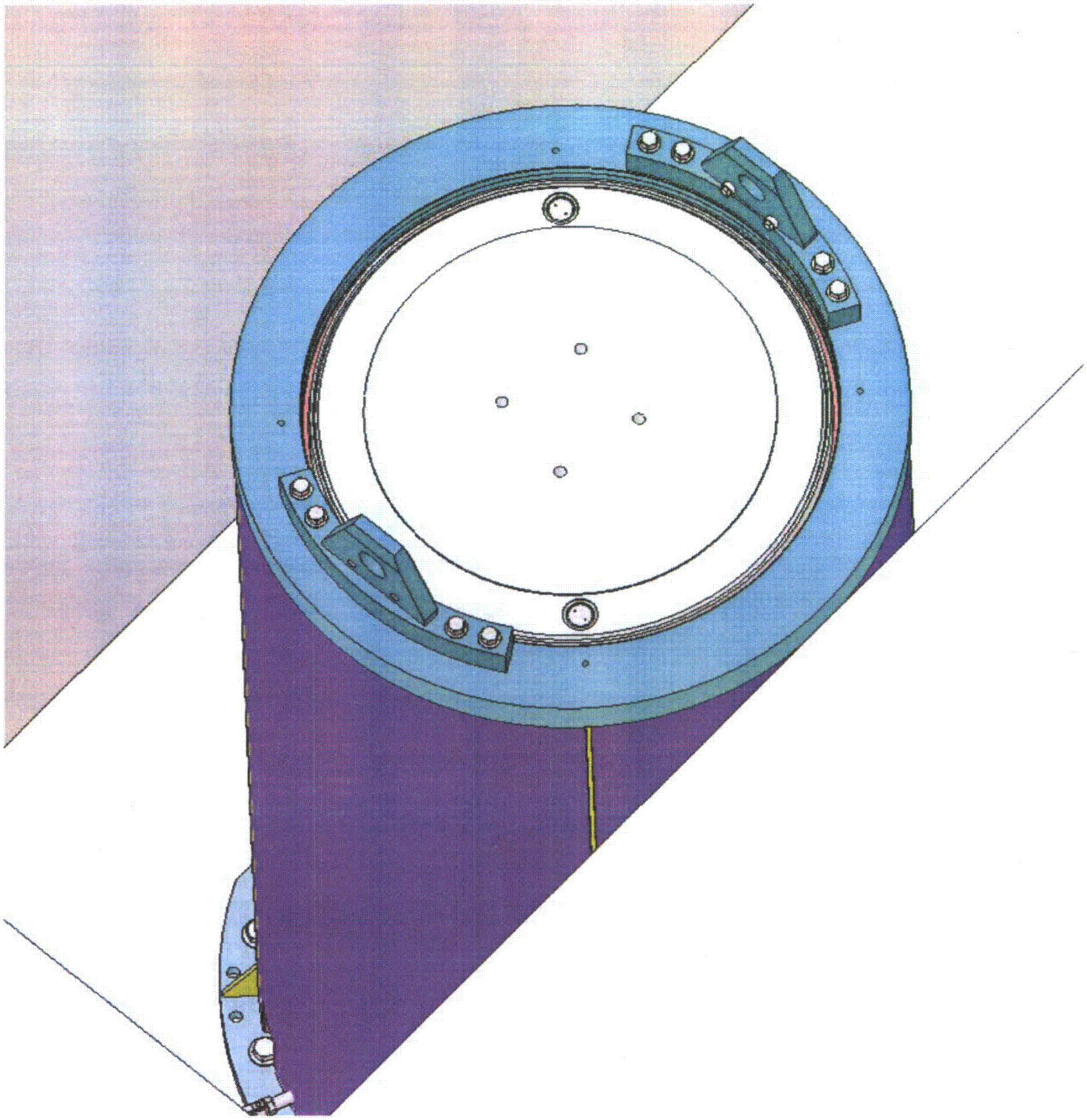

FIGURE 9.2.7: HI-TRAC PLACEMENT IN THE CASK PREPARATION AREA

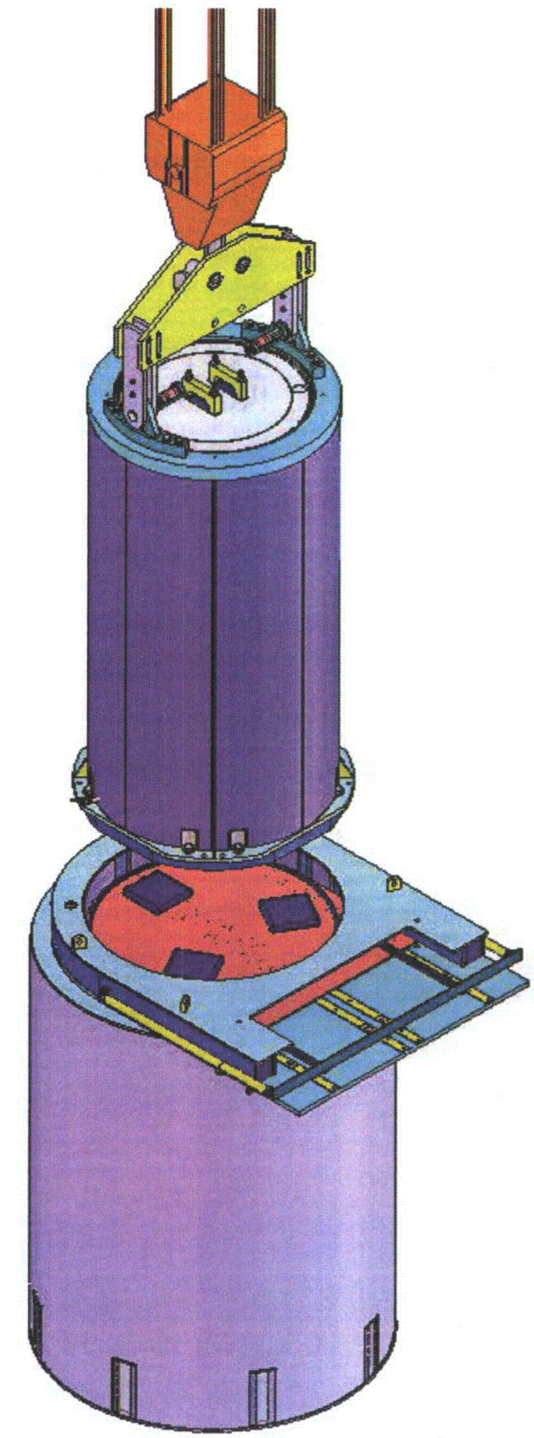

FIGURE 9.2.8: HI-TRAC PLACEMENT ON THE HI-STORM 100 OVERPACK USING THE MATING DEVICE

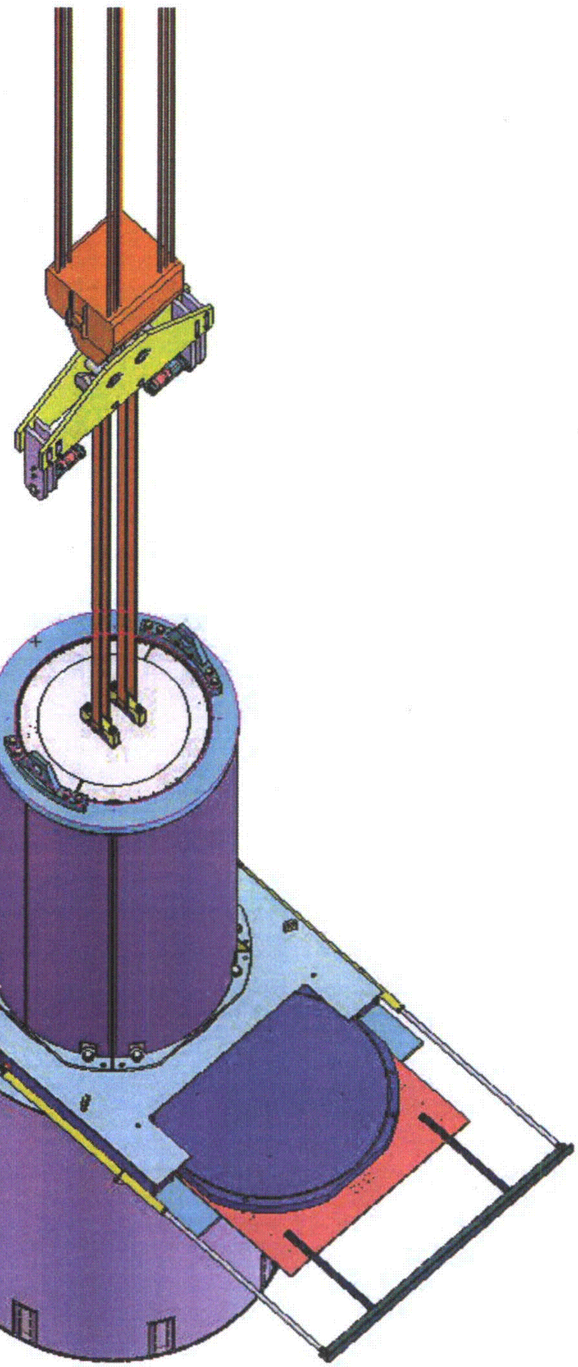

FIGURE 9.2.9: HI-TRAC READY FOR MPC TRANSFER INTO HI-STORM FW OVERPACK

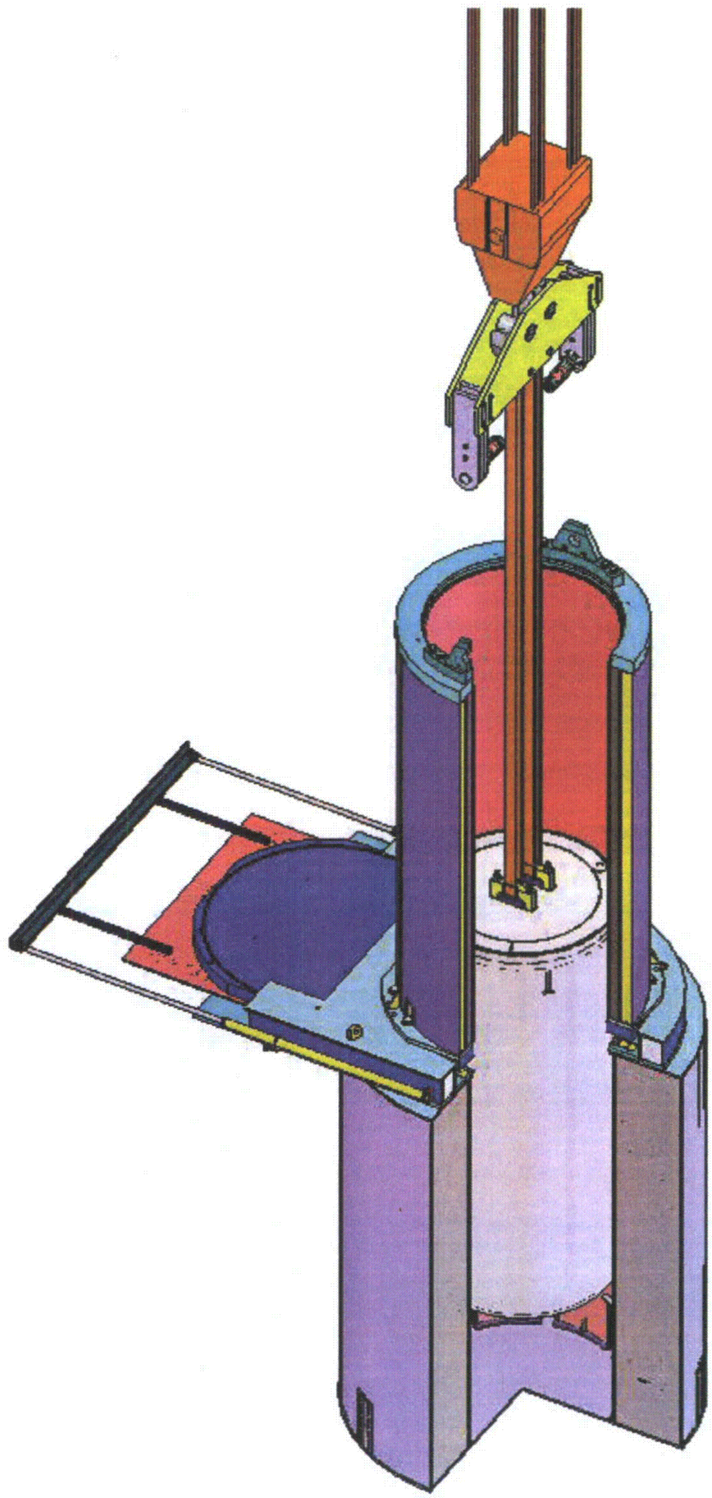

FIGURE 9.2.10: MPC TRANSFER INTO HI-STORM FW OVERPACK (CUT-AWAY VIEW)

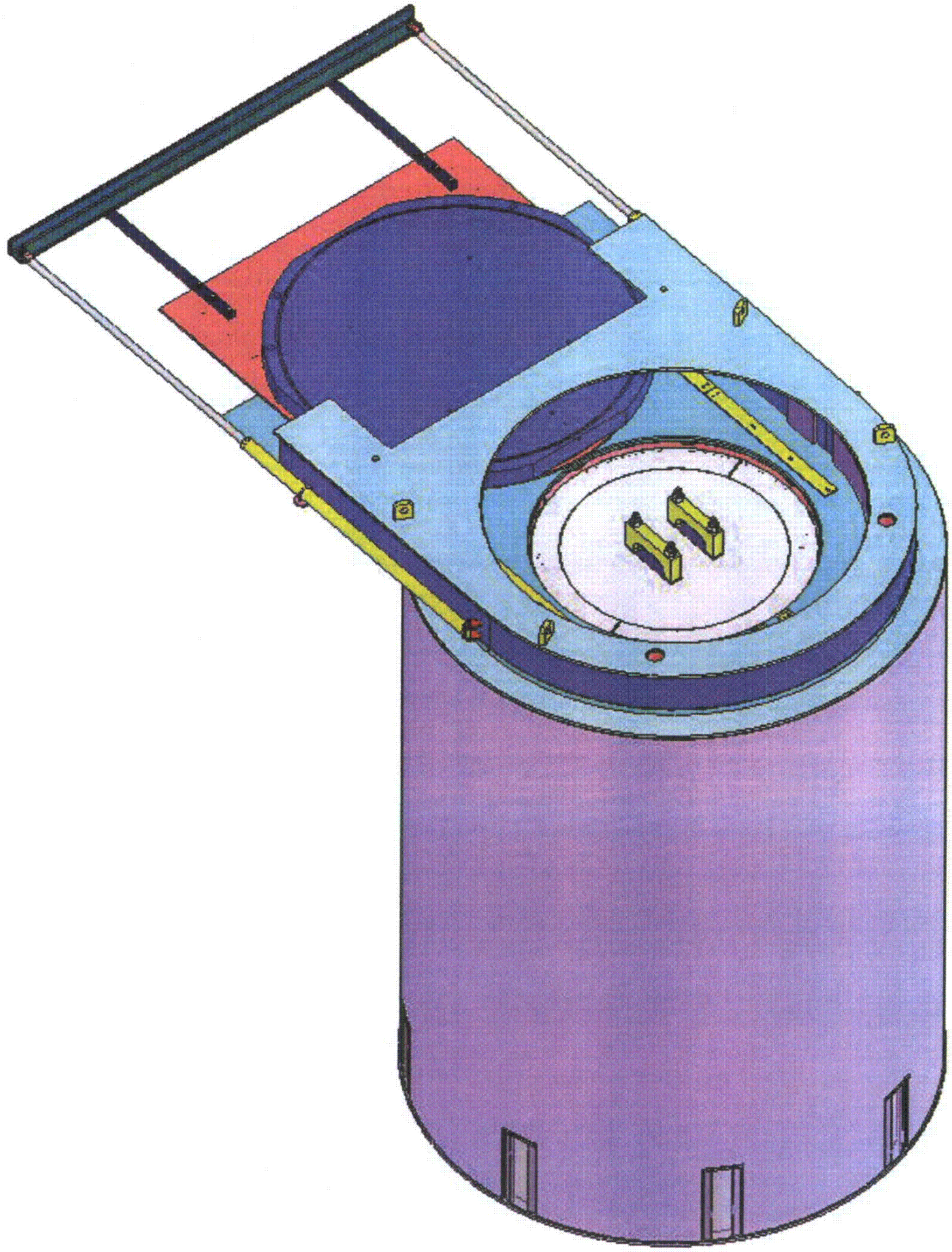

FIGURE 9.2.11: MPC SHOWN FULLY LOWERED INTO HI-STORM (HI-TRAC NOT SHOWN)

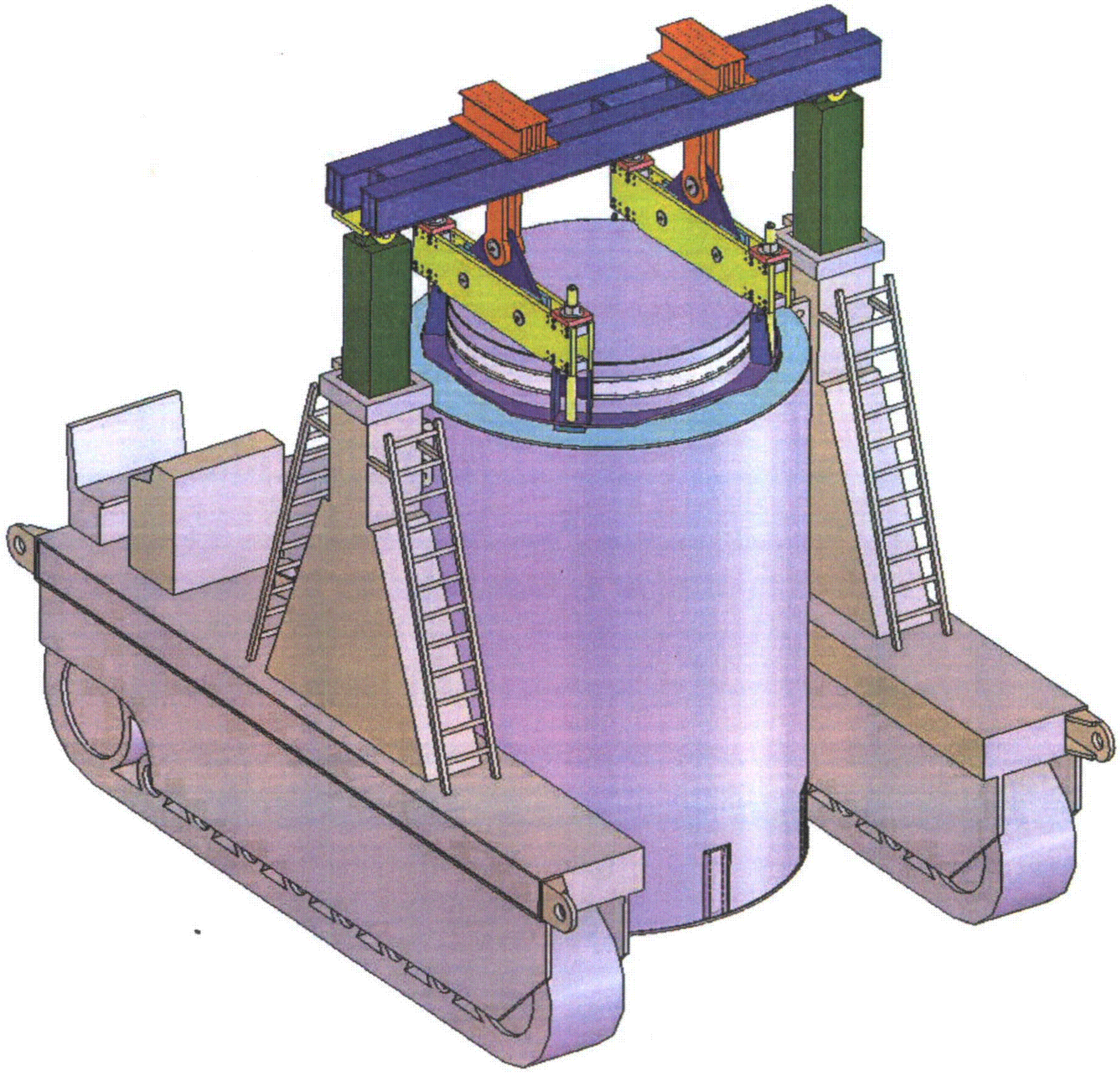

FIGURE 9.2.12: HI-STORM FW OVERPACK MOVEMENT SHOWN WITH A REPRESENTATIVE CASK TRANSPORTER

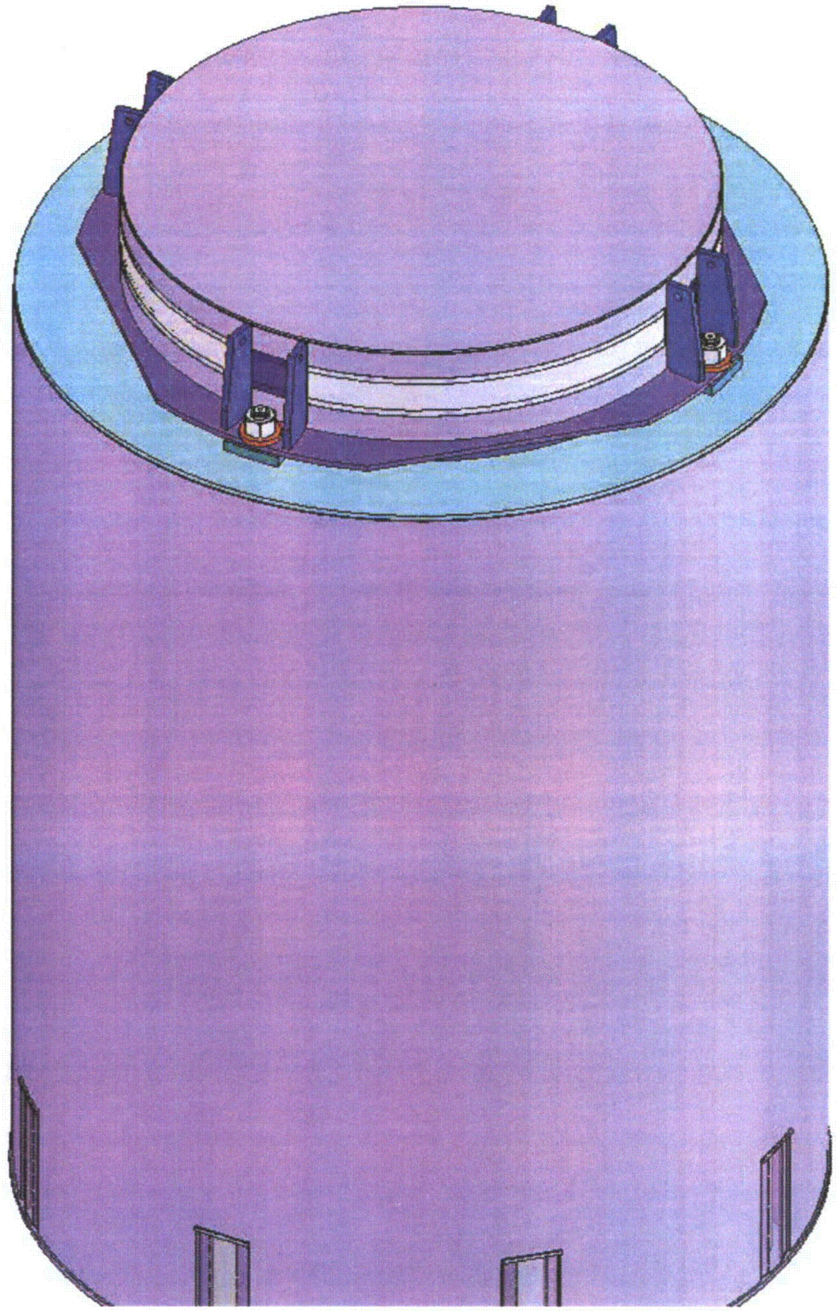

FIGURE 9.2.13: HI-STORM SHOWN IN STORAGE WITH THE LID INSTALLED

9.3 ISFSI OPERATIONS

The HI-STORM FW system heat removal system is a totally passive system. Maintenance on the HI-STORM FW system is typically limited to cleaning and touch-up painting of the overpacks, repair and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM FW system. These include, but are not limited to, wind storms, heavy snow storms, fires inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM FW overpack, clear the debris, check the cask pitch, and/or replace damaged vent screens), perform the corrective action, verify that the system is operable (check ventilation flow paths and radiation). In the unlikely event of significant damage to the HI-STORM FW, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM FW overpack. If necessary, the procedures in Section 9.2 may be used to reposition a HI-STORM FW overpack for minor repairs and maintenance. In extreme cases, Section 9.4 may be used as guidance for unloading the MPC from the HI-STORM FW.

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

ALARA Warning:
Operators should practice ALARA principles when inspecting the vent screens. Binoculars or boroscopes may be used to allow the operator to perform the surveillance from a low dose area.

1. Perform the heat removal operability surveillance in accordance with the CoC.

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

9.4 PROCEDURE FOR UNLOADING THE HI-STORM FW FUEL IN THE SPENT FUEL POOL

9.4.1 Overview of HI-STORM FW System Unloading Operations

ALARA Note:
The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM FW system unloading procedures describe the general actions necessary to prepare the MPC for unloading, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC VW and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. The principal operational steps are summarized below.

The MPC is recovered from HI-STORM FW either at the ISFSI or the fuel building using the same methods as described in Section 9.2 (in reverse order). The HI-STORM FW lid is removed and the mating device is positioned on the HI-STORM FW. MPC slings are attached to the MPC lift attachment and positioned on the MPC lid. HI-TRAC VW is positioned on top of HI-STORM FW and the slings are brought through the top of the HI-TRAC VW. The MPC is raised into HI-TRAC VW, the mating device drawer is closed, and the bottom lid is bolted to the HI-TRAC VW. The HI-TRAC VW is removed from on top of HI-STORM FW.

HI-TRAC VW and its enclosed MPC are returned to the designated preparation area and the MPC lift rigging is removed. Water is added into the annulus space between the MPC and HI-TRAC VW, if required. The annulus and HI-TRAC VW top surfaces are covered to protect them from debris produced when removing the MPC lid weld. The weld removal system is installed and the MPC vent and drain ports are accessed. The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is filled with water (borated as required) at a controlled rate to avoid over-pressuring the MPC. The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place.

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. If weight limitations require, the neutron shield jacket is drained of water. 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 inspection in accordance with a site-specific inspection checklist.

Note:
If the HI-TRAC VW is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

3. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.

4. Engage the lift yoke to HI-TRAC VW.

5. Align HI-TRAC VW over HI-STORM FW and mate the overpacks.

6. Unbolt the bottom lid and open the mating device drawer.

7. Attach the ends of the MPC sling to the lifting device.

8. Raise the MPC into HI-TRAC VW.

9. Verify the MPC is in the full-up position.

10. Close the mating device.

11. Bolt the bottom lid to the HI-TRAC VW.

12. Lower the MPC onto the bottom lid.

13. Disconnect the MPC lift rigging from the MPC lid.

14. Remove HI-TRAC VW from the top of the HI-STORM FW.

9.4.3 Preparation for Unloading

1. Prepare for MPC cool-down as follows:

Warning:
At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus fill. Users may also elect the source of water for the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased.

2. If necessary, set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC VW top surfaces to protect them from debris produced when removing the MPC lid weld.

3. Access the MPC as follows:

ALARA Note:
The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically removed.

ALARA Warning:
Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

a. Using the marked locations of the vent and drain ports, core drill the closure ring and port cover plates.

b. Remove the closure ring sections and the vent and drain port cover plates.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

4. Take an MPC gas sample as follows:

Note:
Users may select alternate methods of obtaining a gas sample.

a. Attach the RVOAs.

b. Attach a sample bottle to the vent port RVOA.

c. Evacuate the RVOA and Sample Bottle.

d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

5. Fill the MPC cavity with water as follows:

Caution:
The MPC interior shall be filled with helium or another suitable inert gas to avoid exposing the fuel to oxidizing agents while at elevated temperatures. Exposing fuel at elevated temperatures to oxidizing agents can lead to deleterious oxidation of the fuel.

a. Open the vent and drain port caps using the RVOAs.

Caution:
The introduction of water into the MPC may create water vapor. Re-flooding operations shall be closely controlled to ensure that the internal pressure in the MPC does not exceed design limits. The water flow rate shall be adjusted to maintain the internal pressure below design limits. See LCO 3.1.3 and SAR section 4.5.5.

Caution:
To mitigate unfavorable thermal shocking of the fuel cladding during re-flooding operations the re-flood water shall be at a temperature ≥80°F. See Section 3.4.4 for related fuel cladding evaluations.

b. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

Note:
When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC. Testing must be completed within four hours prior to unloading and every 48 hours after in accordance with the LCO until all the fuel is removed from the MPC. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

c. Attach the water fill line from a water source with water temperature ≥80°F to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. (Refer to LCO 3.3.1 for boron concentration requirements). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

d. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.

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 removal operations. The space below the MPC lid shall be purged with inert gas prior to, and during MPC lid removal operations, including grinding, and other hot work, to provide additional assurance that flammable gas concentrations will not develop in this space.

e. Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge the gas space under the lid as necessary.

f. Remove the MPC lid-to-shell weld using the weld removal system.

g. Remove any metal shavings from the top surfaces of the MPC and HI-TRAC VW.

6. Install the inflatable annulus seal.
7. Place HI-TRAC VW in the spent fuel pool as follows:

a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.

b. Engage the lift yoke to HI-TRAC VW lifting blocks, remove the MPC lid lifting plugs and attach the MPC lid slings.

c. Position HI-TRAC VW into the spent fuel pool in accordance with site-approved rigging procedures.

d. Disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

e. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel.

f. Disconnect the drain line from the MPC lid.

g. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid.

9.4.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.

2. Remove any debris or corrosion products from the MPC cells.

9.4.5 Post-Unloading Operations

1. Remove HI-TRAC VW and the unloaded MPC from the spent fuel pool as follows:

   a. Engage the lift yoke to the HI-TRAC VW lift blocks.

   b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lift blocks.

   c. Raise HI-TRAC VW until HI-TRAC VW flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of HI-TRAC VW during fuel unloading.

   d. Measure the dose rates at the top of HI-TRAC VW in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

   e. Raise the top of HI-TRAC VW and MPC to the level of the spent fuel pool deck.

   f. Close the annulus overpressure system reservoir valve, if used.

   g. Lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of HI-TRAC VW, the surfaces of HI-TRAC VW and lift yoke should be kept wet until decontamination can begin.

   h. Remove HI-TRAC VW from the spent fuel pool under the direction of radiation protection personnel.

   i. Disconnect the annulus overpressure system from the HI-TRAC VW.

j. Place HI-TRAC VW in the designated preparation area.

k. Disengage the lift yoke.

l. Perform decontamination on HI-TRAC VW and the lift yoke.

m. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

n. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

o. Drain the water in the annulus area by connecting the drain line to the HI-TRAC VW drain connector.

p. Remove the MPC from HI-TRAC VW and decontaminate the MPC as necessary.

q. Decontaminate HI-TRAC VW.

r. Return any HI-STORM FW equipment to storage as necessary.

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".

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

CHAPTER 10†: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

10.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM FW system (overpack, MPC and transfer cask) to verify that the structures, systems and components (SSCs) classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM FW system fabricated, assembled, inspected, tested, and accepted for use under the purview of the HI-STORM FW system CoC.

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters ensure that the HI-STORM FW system will maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions; will maintain subcriticality control; will reject the decay heat of the stored radioactive materials to the environment by passive means and maintain radiation doses within regulatory limits.

Both pre-operational and operational tests and inspections are performed throughout HI-STORM FW system operations to assure that the HI-STORM FW system is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, pressure tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 9 identifies the tests and inspections. "Pre-operation" as referred to in this chapter defines that period of time from receipt inspection of a HI-STORM FW system until the empty MPC is loaded into a HI-TRAC transfer cask for fuel assembly loading.

The HI-STORM FW system is classified as important-to-safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM FW system shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance with a quality program commensurate with the particular SSC's graded quality category. The licensing drawings identify all important to safety subcomponents of the HI-STORM FW system.
The acceptance criteria and maintenance program described in this chapter comply with the requirements of 10CFR72 [10.0.1] and NUREG-1536 [10.0.2] to the maximum extent possible, as described in Chapter 1.

† 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.

The acceptance test requirements on the manufactured welds in the HI-STORM FW system are contained in the component licensing drawings in Section 1.5. Additional details on the requirements in the drawings are provided in this chapter, which will be incorporated in the shop manufacturing documents (viz., weld procedures, shop travelers, inspection procedures, and fabrication procedures) to ensure full compliance with this FSAR.

10.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM FW system prior to and during loading of the system. These inspections and tests provide the assurance that the HI-STORM FW system has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the provisions of 10CFR72 [10.0.1].

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM FW overpack, and the HI-TRAC VW transfer casks are listed in Tables 10.1.1, 10.1.2, and 10.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 9 and 13 provide operating guidance and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM FW system has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR. Identification and resolution of manufacturing noncompliances, if any, shall be performed in accordance with the Holtec International Quality Assurance Program approved by the USNRC.

The contents of this chapter related to welding non-destructive examination are presented in the drawing package in Section 1.5. Likewise, the material on testing and maintenance of system components in this FSAR governs the content of the daughter documents such as the Manufacturing Manual and O&M Manual for the system components used in the manufacturing and long-term maintenance of the system components, respectively.

10.1.1 Fabrication and Nondestructive Examination (NDE)

This subsection summarizes the test program required for the HI-STORM FW system.

10.1.1.1 Fabrication Requirements

The following fabrication controls and required inspections shall be performed on the HI-STORM FW system, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

i. Materials of construction specified for the HI-STORM FW system are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by the ASME Code [10.1.1] Section II (where applicable), the requirements of ASME Section III (where applicable), Holtec procurement specifications, and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to ensure that material traceability is maintained throughout fabrication. Materials for the Confinement Boundary (MPC baseplate, lid, closure ring, port cover plates and shell) shall also be procured in compliance with the requirements of ASME Section III, Article NB-2500.

ii. The MPC Confinement Boundary shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NB to the extent practicable, as explained in this chapter.

iii. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable code (such as ASME Section III Subsection NB for the Confinement Boundary.

iv. Code welds shall be visually examined in accordance with ASME Code, Section V, Article 9. The acceptance criteria for the welds shall be based on the ASME Codes provided in Table 10.1.5. These additional NDE criteria are also specified on the licensing drawings in Section 1.5 for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be subject to review and approval by Holtec in accordance with the Company's QA program prior to use. NDE inspections of code welds shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [10.1.2] or other site-specific, NRC-approved program for personnel qualification.

v. The MPC confinement boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, RT, UT, MT and/or PT, as applicable) to verify that it is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its confinement effectiveness.

vi. Repair of confinement boundary welds shall conform to the requirements of the ASME Code, Section III, Article NB-4450.

vii. Base metal repairs shall be performed and examined in accordance with the applicable reference code set down in Table 10.1.5.

viii. Grinding and machining operations on the MPC Confinement Boundary shall be controlled through written and approved procedures to ensure grinding and machining operations do not reduce local base metal wall thicknesses of the Confinement Boundary below allowable limits. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets the applicable requirements.

ix. Non-structural tack welds that do not become an integral part of a weld are not required to be removed. Non-structural tack welds that do not become an integral part of a permanent weld shall be examined by an approved visual examination procedure.

x. The HI-STORM FW system shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.

xi. Each cask shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

xii. A documentation package shall be prepared and maintained during fabrication of each HI-STORM FW system to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM FW system or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, as applicable, but not be limited to:

- Completed Shop Weld Records
- Inspection Records
- Nonconformance Reports
- Material Test Reports
- NDE Reports
- Dimensional Inspection Report

10.1.1.2 MPC Lid-to-Shell Weld Inspection

i. The MPC lid-to-shell (LTS) weld shall be examined using a progressive multi-layer liquid penetrant (PT) examination during welding.

ii. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed as specified in the drawing package in Section 1.5.

The inspection results, including relevant findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676.

The multi-layer PT examination of the LTS weld, in conjunction with other examinations and tests performed on this weld, shall use ASME Section III acceptance criteria (see Table 10.1.4) which provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The multi-layer PT examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

10.1.1.3 Visual Inspections and Measurements

The HI-STORM FW system components shall be assembled in accordance with the licensing drawing package in Section 1.5. The drawings provide dimensional tolerances that define the limits on the dimensions used in licensing basis analysis. Fabrication drawings provide additional dimensional tolerances necessary to ensure fit-up of parts. Visual inspections and measurements shall be made and controls shall be exercised to ensure that the cask components conform to the dimensions and tolerances specified on the licensing and fabrication drawings. These dimensions are subject to independent confirmation and documentation in accordance with the Holtec QA program approved in NRC Docket No. 71-0784.

The following shall be verified as part of visual inspections and measurements:

- Visual inspections and measurements shall be made to ensure that the systems' effectiveness is not significantly reduced as a result of manufacturing deviations. Any *important-to-safety* component found to be under the specified minimum thickness shall be justified under the rules of 10CFR72.48 or repaired or replaced, as appropriate.
- Visual inspections shall be made to verify that neutron absorber panels, basket shims and anti-rotation bars are present as required by the MPC basket design.
- The system components shall be inspected for cleanliness and preparation for shipping in accordance with written and approved procedures.

The visual inspection and measurement results for the HI-STORM FW system shall become part of the final quality documentation package.

10.1.1.4 Weld Examination

The examination of the HI-STORM FW system welds shall be performed in accordance with the drawing package in Section 1.5 and the applicable codes and standards.

All code weld inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A. All required inspections, examinations, and tests shall become part of the final quality documentation package.

The following specific weld requirements shall be followed in order to verify fabrication in accordance with the provisions of this FSAR.

1. Confinement Boundary welds including any attachment welds (and temporary welds to the Confinement Boundary) shall be examined in accordance with ASME Code Section V, with acceptance criteria per ASME Code Section III, Subsection NB, Article NB-5300. Examinations, Visual (VT), Radiographic (RT), and Liquid Penetrant (PT), apply to these

welds as defined by the code. These welds shall be repaired in accordance with the requirements of the ASME Code Section III, Article NB-4450 and examined after repair in the same manner as the original weld.

2. Basket welds shall be VT examined and repaired in accordance with written and approved procedures developed specifically for welding Metamic-HT and in accordance with the philosophy of ASME Section IX.

3. Non-code welds shall be examined and repaired in accordance with written and approved procedures as defined in the system Manufacturing Manual.

10.1.2 Structural and Pressure Tests

10.1.2.1 Lifting Locations

The HI-STORM FW system does not utilize any lifting trunnions. The lifting of all HI-STORM FW components is engineered to occur though threaded couplings integral to the strongest part in the component. Thus, as shown in the HI-TRAC VW drawings (Section 1.5) the threaded connection is located in the top forging. These lift locations are accordingly referred to as *tapped anchor locations* (TAL). The TALs to lift the MPCs (in all Holtec designs) is located in the top lid (thickest part) and those for the HI-STORM FW overpack are welded to the radial connector plates (in all HI-STORM models).

Because the TALs are integral to the component, they possess high ductility and, as shown in Chapter 3, meet the factor of safety of 6 to yield and 10 to ultimate, as required by ANSI N14.6 [10.1.3].

Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure is one:

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

Each TAL will be subjected to a dimensional test in the shop using go/no-go gauges to ensure that the threads meet the dimensional requirements. As an alternative to the thread gauge test, the threads may be proof-tested using a torque test to simulate a load equal to three times the design load. Furthermore, the thread in the TAL shall be visually inspected in accordance with a written procedure to ensure absence of burrs, undercuts, and other stress raisers.

The acceptance testing of the TALS in the manner described above will provide adequate assurance against handling accidents.

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 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. 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 re-examined by liquid penetrant examination in accordance with ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

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.

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.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as set down in Chapter 1.D of the HI-STORM 100 FSAR (Docket 72-1014) [10.1.6] in accordance with written and approved procedures. Testing shall verify the compressive strength and density meet design requirements. Tests required shall be performed at a frequency as defined in the applicable ACI code.

Qualification tests on Metamic-HT coupons drawn from production runs shall be performed in compliance with Table 10.1.6 requirements to ensure that the manufactured panels shall render their intended function. Testing shall be performed using written and approved procedures consistent with the test methods documented in Holtec's test report [10.1.7].

To ensure the above test requirements are met a sampling plan based on the MIL Standard 105E [10.1.8] is defined and incorporated in the Metamic-HT Manufacturing Manual's Shop Operating Procedure HTSOP-108. The MIL Standard test protocols are selected to maximize coupon population. The test plan has the following key attributes:

- At the start of the Metamic HT production runs at the manufacturing facility, the number of extrusions subjected to testing is based upon the Tier 1 sample population per Table 10.1.7
- If every one of the seven properties meets its respective MGV value given in Table 10.1.8, then the subject lot of extrusions is determined to be acceptable.
- If all tested coupons in five consecutive lots pass (i.e., each of the seven properties in Table 10.1.8 meets its MGV requirement) then the required coupon population can drop down to the Tier 2 sample population per Table 10.1.7.
- If all tested coupons in an additional five consecutive lots pass, then the required coupon population can drop down to the Tier 3 sample population per Table 10.1.7.
- If all tested coupons in an additional ten consecutive lots pass, then the required coupon population can drop down to the Tier 4 sample population per Table 10.1.7.
- If a coupon fails with respect to any property, then it can be replaced by two coupons from the extrusion that produced the failed coupon. If both of the replacement coupons pass all of the seven MGV properties, then the lot can be accepted. If either of the replacement coupons fails to meet any of the seven properties, then the entire lot is rejected.
- The failure of any coupons in a lot requires that subsequent sampling be conducted per the Tier 1 sample population per Table 10.1.7. A reduction to Tier 2, Tier 3, and Tier 4 sample populations in subsequent lots shall be based on the sampling plan described above.

While the above test regimen appears to be quite stringent, the data obtained thus far, and setting the MGVs as smallest of any measured values means that the pull-back to Tier 1 population would be an exception rather than the norm in the production runs.

Test results on all materials shall be documented and become part of the final quality 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 shell and MPC shell to baseplate welds is performed on the unloaded MPC. The acceptance criterion is "leaktight" as defined in ANSI N14.5. 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.

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 shall be performed after welding of the cover plates and subsequent NDE. 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 MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully-welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and are liquid penetrant examined and leakage tested to verify the MPC Confinement Boundary.

There are multiple pressure relief devices installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. One is provided for venting air and water due to pressure build-up from thermal expansion of the water in the water jacket. The other relief devices are provided for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The set pressures for the pressure relief devices are listed on the HI-TRAC VW drawings in Section 1.5.

10.1.5.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM FW system.

10.1.6 Shielding Integrity

The HI-STORM FW overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM FW overpack concrete provides both neutron and gamma shielding. The overpack's inner and outer steel shells, and the steel shield shell, provide radial gamma shielding. Concrete and steel plates provide axial neutron and gamma shielding.

The HI-TRAC VW transfer cask uses three different materials for primary shielding. All HI-TRAC VW transfer cask designs include a radial steel-lead-steel shield and a removable steel bottom lid. Testing requirements on shielding materials are presented below.

Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR drawings in Section 1.5 prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 10.1.3 for concrete material testing requirements.

Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure that voids are minimized. The lead shall be examined to preclude macrovoids in the material using written and qualified procedures.

The lead shall be installed in such a manner that there are no macro-voids and that the cask is not subjected to a severe thermal cycle.

Steel:

Steel plates utilized in the construction of the HI-STORM FW system shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

General Requirements for Shield Materials:

1. Test results for concrete density and lead examinations for macrovoids, as applicable, shall be documented and become part of the quality documentation package.

2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed after initial loading operations in accordance with description below and the operating procedures in Chapter 9.

10.1.6.1 Shielding Effectiveness Tests

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures at the host plant site. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM FW overpack and HI-TRAC VW. Measurements shall be taken at the locations specified in the Radiation Protection Program for comparison against the prescribed limits. The test is considered acceptable if the dose rate readings are less than or equal to the calculated limits. If dose rates are higher than the limits, the required actions provided in the Radiation Protection Program shall be carried out. Dose rate measurements shall be documented and shall become part of the quality record of the loaded cask.

10.1.6.2 Neutron Absorber Manufacturing Requirements

Essential characteristics of Metamic-HT are described in Chapter 1 of this FSAR. As described in Chapter 1, Metamic-HT is made from high purity aluminum using a powder metallurgy process that results in pinning of the materials grain boundaries by dispersoids of nanoparicles of aluminum oxide. The manufacturing of Metamic-HT is governed by a set of quality validated shop procedures contained in a Manufacturing Manual [1.B.2].

The key constituents of Metamic-HT, namely aluminum powder and Boron Carbide powder are procured under their respective purchasing specifications that define the required particle size distributions and set down the prohibited materials & impurities, as well as tolerable level of trace amounts of acceptable impurities.

A description of the manufacturing processes for Metamic-HT is presented in Chapter 1, Appendix 1.B.

As required by the procedures set down in its manufacturing manual [1.B.2], each panel of Metamic-HT neutron absorber material shall be visually inspected for damage such as scratches, cracks, burrs, presence of imbedded foreign materials, voids and discontinuities that could significantly affect its functional effectiveness.

Metamic-HT panels will be manufactured according to a Holtec purchase specification that incorporates all requirements set forth in this FSAR. The supplier of raw materials must be qualified under Holtec's quality program for important to safety materials and components or the material shall be commercially dedicated by Holtec in accordance with the Holtec Quality Assurance program. The manufacturing of Metamic-HT is subject to all quality assurance requirements under Holtec International's NRC approved quality program.

The tests conducted on Metamic-HT to establish the compliance of the manufactured panels with Holtec's Purchasing Specification are intended to ensure that *critical characteristics* of the final product will meet the minimum guaranteed values (MGVs) set forth in this FSAR (Chapter 1, Appendix 1.B). The tests are performed at both the raw material and manufactured extrusion/panel stages of production with the former serving as the insurer of the properties in the final product and the latter serving the confirmatory function. Table 10.1.6 provides a summary of the required tests, their frequency and their intended purpose. The terms "batch" and "lot" referred to in Table 10.1.6 have the following meanings in the context of the manufacturing of Metamic-HT.

- Lot: A lot of B_4C or of aluminum powder is the bulk of material provided by the raw material supplier with a specific property characterization data sheet. A Lot of B_4C or aluminum powder is typically in excess of 5,000 lbs.

- Batch: A batch of B_4C/Al mix is made from a distinct combination of lots of B_4C and aluminum powder. All batches of mix derived from the same distinct combination of lots of B_4C and aluminum are considered "sister" batches.

- A lot of Metamic-HT: A lot of Metamic-HT panels is a set of panels made from one or more sister batches of B_4C/Al.

Note
The text in 10.1.6.3, printed in bold below, is incorporated into the HI-STORM FW CoC by reference (CoC Appendix B, Section 3.2.3) and may not be deleted or altered in any way without prior NRC-approval via CoC amendment. The affected verbiage is, therefore, shown in bold type to distinguish it from other text.

10.1.6.3 Neutron Absorber Manufacturing and Tests

Each plate of neutron absorber shall be visually inspected for damage such as scratches, cracks, burrs, peeled cladding, foreign material embedded in the surfaces, voids, delamination, and surface finish, as applicable.

NUREG/CR-5661 identifies the main reason for a penalty in the neutron absorber B-10 density as the potential of neutron streaming due to non-uniformities in the neutron absorber, and recommends comprehensive acceptance tests to verify the presence and uniformity of the neutron absorber for credits more than 75%. Since a 90% credit is taken for Metamic-HT®, the following criteria must be satisfied:

- The boron carbide powder used in the manufacturing of Metamic-HT® must have sufficient fine particle size distribution to preclude neutron streaming.
- The B_4C weight percent shall be 10% (minimum).
- The B_4C powder must be uniformly dispersed locally, i.e., must not show any particle agglomeration. This precludes neutron streaming.
- The B_4C powder must be uniformly dispersed macroscopically, i.e., must have a consistent concentration throughout the entire neutron absorber panel.

To ensure that the above requirements are met the following tests shall be performed (see Table 10.1.6):

- **All lots of boron carbide powder shall be analyzed to meet particle size distribution requirements.**
- **All lots of B_4C will be certified as containing Boron with a minimum 18.3% of isotopic B-10 per the purchase specification.**
- **Wet chemistry testing of a sample from each mixed batch shall be performed to verify the correct boron carbide weight percent of 10% is attained. The mixing of the batch is controlled via approved procedures.**

- **The thickness of each final panel will be measured in at least six places, with two at one end, two at the other end and two in the middle, and shown to meet the minimum basket wall thickness.**

The measurements of B_4C content, particle size, thickness, and uniformity of B_4C distribution (via wet chemistry test) shall be made using written and approved procedures. If any one of the above criterion is not met, the panel will not be used. If the wet chemistry results for a mixed batch do not meet the criteria, all panels from the entire mixed batch will not be used. This ensures the required B_4C content of the Metamic-HT panels is achieved.

As additional verification one coupon from each lot shall be tested via neutron attenuation testing to verify the expected B_4C content is attained. The neutron attenuation testing will be performed using a 1 inch diameter thermal neutron beam that is calibrated using a solid B_4C plate, and the results will be compared to a known standard whose B_4C content has been checked and verified. This test shall be performed to verify the continued acceptability of the manufacturing process. The B_4C content attained by the neutron attenuation tests will be compared to the wet chemistry results. If a coupon fails the neutron attenuation test, all panels from this lot will be rejected.

Each plate of neutron absorber shall be visually inspected for damage such as scratches, cracks, burrs, peeled cladding, foreign material embedded in the surfaces, voids, and delaminations. Panels are also visually inspected for contamination on the surface. Panels not meeting the acceptance criteria will be rejected. Panels are inspected before being shipped to the cask manufacturing facility and they are subject to an additional receipt inspection prior to installation.

10.1.7 Thermal Acceptance Tests

The thermal performance of the HI-STORM FW system, including the MPCs and HI-TRAC transfer cask, is demonstrated through analysis in Chapter 4 of this FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading.

The first manufactured MPC, either MPC-37 or MPC-89, will be thermally tested using an approved QA controlled Holtec procedure [10.1.9]. The following are the basic steps of this procedure.

1. The MPC will be arrayed in the vertical orientation on the test pad with interface insulation to minimize heat loss from the bottom.
2. Twelve storage cells (three in each quadrant) will be loaded with bayonet electric heaters each calibrated to deliver one kilowatt heat uniformly over its length. The heaters will be situated co-axially within each storage cell. Thus the heat generation in the MPC shall be quadrant-symmetric.
3. The top of the MPC shall be enclosed by an insulated lid. Calibrated thermocouples will be fastened to selected cell walls in each quadrant in a symmetric manner.

4. The test will be run for a sufficiently long time such that steady state conditions are reached. The ambient temperature and the thermocouple readings will be taken as specified in the test procedure.
5. The test condition will be simulated on the design basis FLUENT model of the MPC in Chapter 4 and the temperatures at all of the thermocouple locations predicted by FLUENT will be compared with the test data.
6. The amounts by which the FLUENT temperatures exceed the corresponding measured temperatures (positive margin) collectively define the margin of conservatism in the FSAR analysis model. A negative margin will warrant an immediate report to the NRC and appropriate licensing action pursuant to Holtec's QA program.

Following the loading and placement on the storage pad of the first HI-STORM system placed in service as specified in CoC Condition #8, the operability of the natural convective cooling of the HI-STORM FW system shall be verified by the performance of an air mass flow rate test. A description of the test is described in Chapter 9.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM FW heat removal system.

10.1.8 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in 10 CFR 72.236(k).

Table 10.1.1
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of MPC 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 TEST 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 shell and MPC shell to baseplate 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. 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 by inspection.	None.	None.
Shielding Integrity	a) Material compliance is verified through CMTRs. b) Dimensional verification of MPC lid thickness is performed.	None.	None.

Table 10.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Thermal Acceptance	a)None.	a) None.	a) None.
Fit-Up Tests	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 Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 10.1.2 HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE)	Structural Steel Components: a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All structural weld s requiring PT examination as shown on the Licensing Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All structural welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using ANSI NDE symbols and/or notations. Concrete Components: The following processes related to concrete components shall be implemented in accordance with the provisions of Appendix 1.D of [10.1.6]. Concrete testing shall be in accordance with Table 1.D.1. Activities shall be conducted in accordance with written and approved procedures. a) Assembly and examination. b) Mixing, pouring, and testing.	a) The overpack shall be visually inspected prior to placement in service. b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means. c) overpack protection at the licensee's facility shall be verified. d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.	a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned. b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.

Table 10.1.2 (continued) HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. b) Concrete compressive strength tests shall be performed per Appendix 1.D of [10.1.6].	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per Appendix 1.D of [10.1.6], at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement.	a) None	a) A shielding effectiveness test shall be performed after the initial fuel loading.

Table 10.1.2 (continued) HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM FW system in accordance with the operating procedures in Chapter 9. b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's option. (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions. (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading.	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.

Table 10.1.3

HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All structural welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All structural welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using standard ANSI NDE symbols and/or notations e) Cleanliness of the transfer cask shall be verified upon completion of fabrication. f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.	a) The transfer cask shall be visually inspected prior to placement in service. b) Transfer cask protection at the licensee's facility shall be verified. c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.	a) Visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements.

Table 10.1.3 (continued) HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. b) A pressure test of the neutron shield water jacket shall be performed upon completion of fabrication.	a) None.	None.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) None.	a) None.	a) None.
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading.	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask bottom lid shall be performed during fabrication.	None.	a) Fit-up of the bottom lid shall be verified prior to use.

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	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
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

Table 10.1.5

REFERENCE ASME CODES FOR CODE WELD INSPECTIONS AND INSPECTION CRITERIA OF HI-STORM FW COMPONENTS

Component	Applicable Reference Code for Inspection Criteria	Applicable Code for Inspection Process
MPC Confinement Boundary	ASME Section III Subsection NB	Section V
HI-STORM FW Overpack Steel Weldment	ASME Section III Subsection NF for Class 3 Structures	Section V
HI-TRAC VW Transfer Cask (Steel Weldment)	ASME Section III Subsection NF for Class 3 Structures	Section V

	Table 10.1.6 Metamic-HT Testing Requirements				
	Item Tested	**Property Tested For**	**Frequency of Test**	**Purpose of Test**	**Acceptance Criterion**
i.	B_4C powder (raw material) (see note 1)	Particle size distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification
		Purity	One sample per lot	To verify material supplier's data sheet	ASTM C-750
ii.	Al Powder (raw material)	Particle Size Distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification
		Purity	One sample per lot	To verify material supplier's data sheet	Must be 99% (min.) pure aluminum
iii.	B_4C/Al Mix	B_4C Content (by the wet chemistry method)	One sample per batch	To ensure wt.% B_4C requirements compliance	The weight density of B_4C must lie in the range specified in the drawing package in Section 1.5.
iv.	Finished Metamic-HT panel	Thickness and width, straightness, camber and bow	Each Panel	To ensure fabricability of the basket	Per Holtec's Purchasing Specification
		Mechanical & Impact Properties, See Table 10.1.8.	Per Sampling Plan (see Note 2)	To ensure structural performance.	MGV per Table 10.1.8
		B-10 areal density (by neutron attenuation)	One coupon from each Metamic-HT manufactured lot	To ensure criticality safety	The weight density of B_4C must lie in the range specified in the drawing package in Section 1.5.
		Thermal Conductivity	One Sample from each Metamic-HT manufactured lot	To ensure thermal performance	MGV per Table 1.B.2

Notes:

1. The B_4C testing requirements apply if the raw material supplier is not in Holtec's (Or Nanotec's) Approved Vendor List.
2. Sampling Plan is included in the Metamic-HT Manufacturing Manual [1.B.2].

Table 10.1.7 Tier System for Coupon Testing		
Tier No.	Number of Extrusions Tested as a Percent of Number of Extrusions in the Lot	Number of Continuous Lots that Must Pass to Drop Down to the Next Tier
1	20	5
2	12.5	5
3	5	10
4	1	N/A

Table 10.1.8 Minimum Guaranteed Values Required for Certification of Production Runs of Metamic-HT (All testing performed at ambient temperature.)		
	Property	MGV
1	Yield Strength, ksi	See Table 1.B.2 for MGV values
2	Tensile Strength, ksi	
3	Young's Modulus, ksi	
4	Elongation, %	
5	Charpy Impact Strength, ft-lb	
6	Lateral Expansion, mils	
7	Area Reduction, %	

10.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM FW system Operations and Maintenance Manual, which shall be prepared and issued prior to the first use of the system by a user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal, and confinement performance, radiological safety, and proper handling of the system in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM FW system is totally passive by design: There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from the effects of weather. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures that are far less demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STORM FW system is provided in Table 10.2.1.

10.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC TALs and the bottom lid bolts and bolt holes. The examination shall inspect for indications of overstress such as cracks, deformation, wear marks, and missing or damaged threads. Repairs or replacement in accordance with written and approved procedures shall be required if an unacceptable condition is identified.

As described in Chapters 7 and 12 of this FSAR, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

10.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 12, there are no credible normal, off-normal, or accident events which can cause the failure of the MPC Confinement Boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

10.2.3 Subsystem Maintenance

The HI-STORM FW system does not include any subsystems, which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, forced helium drying, helium backfill, and leakage testing systems per their O&M manuals. Rigging, remote welders, cranes, and lifting beams shall also be inspected prior to each loading campaign to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations with the HI-STORM FW require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also Subsection 10.2.6.

10.2.4 Pressure Relief Devices

The pressure relief devices used on the water jackets for the HI-TRAC VW transfer cask shall be calibrated as specified in the HI-TRAC VW O&M Manual to ensure pressure relief settings are accurate prior to the cask's use.

10.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM FW overpack, HI-TRAC VW, and MPC are not subject to measurable degradation over time or as a result of usage.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.104(c) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC VW water jacket shall be verified during each loading campaign in accordance with the licensee's approved operations procedures.

The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM FW system.

10.2.6 Thermal

In order to assure that the HI-STORM FW system continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 10.2.1

HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	Frequency
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM FW Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC TAL visual inspection	Prior to each handling campaign
HI-TRAC bottom lid bolts and bolt holes	Prior to each handling campaign
HI-TRAC pressure relief device calibration	Per the device manufacturer's recommendation.
HI-TRAC internal and external visual inspection for compliance with design drawings	Annually
HI-TRAC water jacket water level visual examination	During each handling campaign in accordance with licensee approved operations procedures
Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

10.3 REGULATORY COMPLIANCE

Chapter 10 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STORM FW system in conformance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STORM FW system will provide the assurance that the SSCs important to safety will perform their intended function without limitation. The performance of the maintenance program by the licensee for each loaded HI-STORM FW system will provide the assurance for the continued safe long-term storage of the stored SNF.

The described acceptance criteria and maintenance programs can be summarized in the following evaluation statements:

1. Section 10.1 of this FSAR describes Holtec International's proposed program for pre-operational testing and initial operations of the HI-STORM FW system. Section 10.2 describes the proposed HI-STORM FW system's maintenance program.

2. Structures, systems, and components (SSCs) of the HI-STORM FW system designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with their safety category. The licensing drawings in Section 1.5 and Table 9.2.1 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STORM FW system and its ancillary equipment, respectively. Tables 1.2.6, 1.2.7, and 1.2.8 present the applicable standards for their design, fabrication, and inspection of the HI-STORM FW system components.

3. Holtec International will examine and test the HI-STORM FW system to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 10.1 of this FSAR describes the MPC Confinement Boundary assembly, inspection, and testing.

4. Each cask shall bear a nameplate indicating its model number, unique identification number, and empty weight.

5. It can be concluded that the acceptance tests and maintenance program for the HI-STORM FW system are in compliance with 10CFR72 [10.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the overarching regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

10.4 REFERENCES

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

[10.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.

[10.1.1] American Society of Mechanical Engineers, " Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 2007 Edition.

[10.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.

[10.1.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.

[10.1.4] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.

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

[10.1.6] "Final Safety Analysis Report for HI-STORM 100 Cask Storage System", Holtec Report No. HI-2002444 (latest revision).

[10.1.7] "Metamic-HT Qualification Sourcebook", by I. Rampall, T.G. Haynes, and J. Menhart, Holtec Report No. HI-2084122, (2009) (Holtec Proprietary)[1]

[10.1.8] "Sampling Procedures and Tables for Inspection by Attributes", Military Standard MIL-STD-105E, (10/5/1989).

[10.1.9] "HI-STORM FW MPC Thermal Test Procedure", Holtec Procedure HPP-5018-1, Rev. 0.

[1]Supporting document submitted with the HI-STAR 180 License Application (Docket 71-9325).

CHAPTER 11: RADIATION PROTECTION†

11.0 INTRODUCTION

This chapter discusses the design considerations and operational features that are incorporated in the HI-STORM FW system design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, transfer, and on-site dry storage. Occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections are provided. An off-site dose assessment for a typical ISFSI is also presented. Since the determination of off-site doses is necessarily site-specific, similar dose assessments shall be prepared by the licensee, as part of implementing the HI-STORM FW system in accordance with 10CFR72.212 [11.0.1]. The information provided in this chapter meets the requirements of NUREG-1536 [11.0.3].

11.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

11.1.1 Policy Considerations

The HI-STORM FW has been designed in accordance with 10CFR72 [11.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [11.1.1] and the guidance provided in Regulatory Guides 8.8 [11.1.2] and 8.10 [11.1.3]. Licensees using the HI-STORM FW system will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [11.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM FW system, and be familiarized with the expected dose rates around the MPC, HI-STORM overpack and HI-TRAC VW during all phases of loading, storage, and unloading operations. Chapter 13 provides dose rate limits at the HI-TRAC VW and HI-STORM overpack surfaces to ensure that the HI-STORM FW system is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings will be held with workers and radiological protection personnel prior to work on or around the system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [11.1.1] standards for radiation protection are met in accordance with the site's written commitments.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61[11.0.2]. However, the material content of this chapter also fulfills the requirements of NUREG 1536[11.0.3]. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1 in this SAR. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

11.1.2 Radiation Exposure Criteria

The radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM FW system are as follows:

1. 10CFR72.104 [11.0.1] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements. As discussed below, the design features of the HI-STORM FW system components are configured to meeting this and other criteria cited below without undue burden to the user (discussed in Subsection 11.1.2).

2. 10CFR72.106 [11.0.1] requires that any individual located on or beyond the nearest owner-controlled area boundary may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.

3. 10CFR20 [11.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.

4. Regulatory Position 2 of Regulatory Guide 8.8 [11.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM FW storage system as described below:

   - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [11.0.1]. Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a loaded HI-STORM FW overpack is placed in an ISFSI. Due to the passive nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.

- Regulatory Position 2b, regarding radiation shielding, is met by the storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 and in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM FW system design include:
  - system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;
  - system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;
  - system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;
  - system designs that minimize planned maintenance requirements;
  - system designs that minimize decontamination requirements at ISFSI decommissioning;
  - system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;
  - thick walled overpack that provides gamma and neutron shielding;
  - thick MPC lid which provides effective shielding for operators during MPC loading and unloading operations;
  - multiple welded barriers to confine radionuclides;
  - smooth surfaces (that come in contact with pool water) to reduce decontamination time;
  - minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
  - capability of maintaining uncontaminated water in the MPC during welding to reduce dose rates;
  - capability of maintaining water in the transfer cask annulus space and water jacket to reduce dose rates during closure operations;
  - MPC penetrations located and configured to reduce neutron streaming paths;
  - elimination of trunnions in the HI-TRAC VW, which serve as streaming paths;
  - streaming paths in the HI-STORM FW overpack are limited to the air vent passages.

- MPC vent and drain ports with resealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;
- use of a bottom lid, annulus seal, and Annulus Overpressure System to prevent contamination of the MPC shell outer surfaces during in-pool activities;
- maximization of shielding around the top region of HI-TRAC VW where the most human activities occur during loading operations; and
- low-maintenance design to reduce occupational dose during long-term storage.

- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.
- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM FW storage system is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this SAR, and no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean water in the HI-TRAC VW-MPC annulus and by using a proven inflatable annulus seal design.
- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM FW system ISFSI since there are no radioactive systems at an ISFSI that could transport crud.
- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded transfer cask is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the HI-TRAC VW transfer cask is designed for ease of decontamination. In addition, an inflatable annulus seal is used to prevent fuel pool water from contacting and contaminating the exterior surface of the MPC.
- Regulatory Position 2g, regarding monitoring of airborne radioactivity, is met since the MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI.
- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins.
- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC Enclosure Vessel. This material is resistant to the damaging effects of radiation and is well proven in the SNF cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

11.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STORM FW system include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated welding system, lift yoke, mating device and moisture removal systems to reduce time operators spend in the vicinity of the loaded MPC;
- use of a well-shielded base for staging the welding system;
- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;
- low fuel assembly lift-over height over the HI-TRAC VW maximizes water coverage over assemblies during fuel assembly loading;
- a water-filled neutron shield jacket allows filling after removal of the HI-TRAC VW from the spent fuel pool. This maximizes the shielding on the HI-TRAC VW without exceeding the crane capacity;
- descriptive operating procedures that provide guidance to reduce equipment contamination, obtain survey information, minimize dose and alert workers to possible changing radiological conditions;
- preparation and inspection of the HI-STORM FW overpack and HI-TRAC VW in low-dose areas;
- MPC lid fit tests and inspections prior to actual loading to ensure smooth operation during loading;
- gas sampling of the MPC and HI-STAR 100 annulus (receiving from transport) to assess the condition of the cladding and MPC Confinement Boundary;
- HI-STORM FW overpack temperature monitoring equipment allows remote monitoring of the vent operability surveillance;
- Use of proven ALARA measures such as wetting of component surfaces prior to placement in the spent fuel pool to reduce the need for decontamination;
- decontamination practices which consider the effects of weeping during HI-TRAC VW transfer cask heat up and surveying of HI-TRAC VW prior to removal from the fuel handling building;

- Use of non-porous neutron absorber (Metamic-HT) to preclude waterlogging of the neutron absorber to minimize basket drying time. Specifically, Boral (a sandwich of aluminum sheets containing a mixture of boron carbide and aluminum powder which tends to hold the pool water in the porous space of the mixture extending canister drying times) is prohibited from use in HI-STORM FW MPCs);
- a sequence of short-term operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations

11.1.4 Auxiliary/Temporary Shielding

In addition to the design and operational features built into the HI-STORM FW system components, a number of ancillary shielding devices can be deployed to mitigate occupational dose. Ancillaries are developed on a site-specific basis that further reduce radiation at key work locations and/or allow for operations to be performed faster to reduce the time personnel spend in close proximity in the radiation field. Licensees are encouraged to use such ALARA-friendly ancillaries and practices.

11.2 RADIATION PROTECTION FEATURES IN THE SYSTEM DESIGN

The design of the HI-STORM FW 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 11.2.1. The design measures listed in Table 11.2.1 have been incorporated in the HI-STORM FW system to effectively reduce dose in fuel storage applications.

Table 11.2.1

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 Overpack	Use of the steel weldment structure permits the density of concrete (set at a minimum of 150 lb/cubic feet) to be increased to as high as 200 lb/cubic feet.	A
2.	HI-STORM FW Overpack	The lid of the HI-STORM FW overpack contains the outlet ventilation ducts (Holtec Patent No. 6,064,710) in the overpack's closure lid. This eliminates the need for temporary shielding that will otherwise be needed if the ducts were located in the cask body for MPC transfer operations.	B
3.	HI-STORM FW Overpack	Use of multiple curved inlet ducts maximize radiation blockage (Holtec Patent No. 6,519,307B1).	B
4.	HI-STORM FW Overpack	Cask's vertical disposition and use of a thick lid (see drawing package in Section 1.5) and high density concrete minimizes skyshine.	B
5.	HI-TRAC VW/ MPC	The height of the MPC minimized for each site so that the height of HI-TRAC VW can be minimized and thus the maximum amount of lateral shielding in the cask can be incorporated consistent with the plant's crane capacity limits.	B
6.	HI-TRAC VW	Lifting trunnions located in the upper region of the transfer cask serve as neutron streaming paths in the space where human activity is necessary (welding, NDE, etc.). Eliminating trunnions and replacing them with TALs (see Glossary) eliminates steaming and aids ALARA during operations at the DAS.	B
7.	MPC	Use of Metamic-HT in the fuel basket reduces the weight of the fuel basket (in comparison to stainless steel). Thus additional shielding can be incorporated in the transfer cask whose total weight is limited by the plant's crane capacity.	B

Table 11.2.1

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
8.	HI-STORM FW overpack/MPC	The dose from a HI-STORM FW storage system is minimized because of the following advantages: a. Regionalized storage of fuel (cold fuel in the peripheral storage cells) possible because of the Metamic-HT fuel basket and the thermosiphon action-enabled MPC provides self-shielding. b. Tight packing of overpacks on the ISFSI (that maximizes self-shielding) is possible because a large spacing between the modules is not necessary.	A,B
9.	MPC, HI-TRAC VW	The occupational dose from loading a HI-STORM FW overpack is minimized because of: a. A well-shielded HI-TRAC VW transfer cask. b. Regionalized fuel loading. c. A short water draining time (less than 2 hours) for the MPC. d. Reduced overall MPC welding time because the welding machine does not have to be removed and replaced to weld the secondary lid. e. Reduced time and personnel needed to install the MPC in the HI-STORM FW overpack due to vertical (gravity-aided) insertion. f. Reduced drying time because of use of porosity-free Metamic-HT.	B
10.	MPC	HI-STORM FW has been designed to accommodate high burnup and a maximum number of PWR or BWR fuel assemblies in each MPC to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site.	A,B

Table 11.2.1

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
11.	HI-STORM FW overpack	HI-STORM FW 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
12.	MPC	HI-STORM FW has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the Confinement Boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or credible accident conditions.	A,B
13.	HI-TRAC VW	HI-TRAC VW transfer cask utilizes a mating device (Holtec Patent No. 6,625,246) which reduces streaming paths and simplifies operations.	B
14.	HI-TRAC VW	The HI-TRAC VW cask and mating device are designed for quick alignment with HI-STORM.	B
15.	HI-STORM FW overpack	HI-STORM FW 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
16.	HI-STORM FW overpack	The HI-STORM FW overpack features narrow and tall optimized inlet duct shapes to minimize radiation streaming.	A

Table 11.2.1

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
17.	HI-STORM FW overpack/MPC	The combination of a Metamic-HT (highly conductive) basket, a thermosiphon capable internal basket geometry, and a high profile inlet ducts enables the HI-STORM FW system to reject heat to the ambient to maintain the fuel cladding temperature below short-term limits in the scenario where the ISFSI is flooded and the floodwaters are just high enough to block off the ventilation airflow. This feature eliminates the need for human intervention to protect the fuel from damage from an adverse flood event and reduces occupational dose.	A,B
18.	HI-STORM FW overpack	The steel structure of the HI-STORM FW 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

11.3 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 system. This section uses the shielding analysis provided in Chapter 5, the operations procedures provided in Chapter 9 and the experience from the loading of many MPCs to develop a realistic estimate of the occupational dose.

The dose rates from the HI-STORM FW overpack, MPC lid, HI-TRAC VW, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the fuel loading and unloading operations. No assessment is made with respect to background radiation since background radiation can vary significantly by site.

The estimated occupational dose is governed by three principal parameters, namely:

i. The dose rate emanating from the MPC.
ii. Average duration of human activity in the radiation elevated space.
iii. Relative proximity of humans to the radiation source.

The dose rate accreted by the MPC depends on its contents. Regionalized storage has been made mandatory in the HI-STORM FW MPC to reduce its net radiation output. The duration of required human activity and the required human proximity, on the other hand, are dependent on the training level of the personnel, and user friendliness of ancillary equipment and the quality of fit-up of parts that need to be assembled in the radiation field.

To provide a uniform basis for the dose estimates presented in this chapter, the reference MPC contents data, available HI-TRAC VW weight, etc., are set down in Table 11.3.1.

Using Table 11.3.1 data, the dose data for fuel loading (wet to dry storage) is provided in Table 11.3.2. The dose for the reverse operation (dry to wet storage) is summarized in Table 11.3.3.

For each step in Table 11.3.2, the task description, average number of personnel in direct radiation field, exposure duration in direct radiation field and average dose rate are identified. The relative locations refer to all HI-STORM FW overpacks. The dose rate location points around the transfer cask and overpack were selected based on actual experience in loading HI-STORM 100 Overpacks. Cask operators typically work with workers entering and exiting the immediate cask area. To account for this, an average number of workers and average dose rates are used. The tasks involved in each step presented in Table 11.3.2 are not provided in any specific order.

11.3.1 Estimated Exposures for Loading and Unloading Operations

Exposures estimates presented in Tables 11.3.2 is expected to bound those for unloading operations. This assessment is based on the similarity of many loading versus 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.

11.3.2 Estimated Exposures for Surveillance and Maintenance

Table 11.3.4 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM FW modules or remote viewing methods instead of performing direct visual observation of the modules. Since security surveillances can be performed from outside the ISFSI, and since the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area, a dose rate of 3 mrem/hour is estimated. Although the HI-STORM FW system requires only minimal maintenance during storage (e.g., touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Since most of the maintenance is expected to occur outside the actual cask array, a dose rate of 10 mrem/hour is estimated.

Table 11.3.1 ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE		
	Item	**Value**
1.	**MPC-Contents (MPC-37)[1]**	**45,000 MWD/MTU and 5 years**
2.	**Weight of HI-TRAC VW Full of Fuel and Water**	**125 tons**
3.	**HI-STORM Concrete Density**	**150 lb/cubic feet**

[1] The case of MPC-37 is used but similar results are expected for the MPC-89.

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average 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	40
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	45	68
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	45	28

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average 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	67	76
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	200	53
HI-STORM FW system preparation for receiving MPC (includes: HI-STORM FW overpack positioning at transfer location, HI-STORM lid removal, Mating Device installation on HI-STORM FW overpack).	3	160	0	0

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average 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	100	60
HI-STORM FW 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	25	55
TOTAL EXPOSURE (person-mrem)	380			

Table 11.3.3

ESTIMATED EXPOSURES FOR HI-STORM FW SURVEILLANCE AND MAINTENANCE

ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 11.3.2, 11.3.3, AND 11.3.4:

1. Refer to Chapter 9 for detailed description of activities.
2 Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.

11.4 ESTIMATED CONTROLLED AREA BOUNDARY DOSE ASSESSMENT

11.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [11.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STORM FW system with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site-specific dose analysis for their particular situation in accordance with 10CFR72.212 [11.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.3 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [11.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. Table 11.4.1 provides the annual dose results for a single HI-STORM FW under the conditions specified in Table 11.3.1. 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 indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of 300 meters for a single cask and at 500 meters for the cask group of HI-STORM FW systems containing Table 11.3.1 fuel. These results are presented only as an illustration to demonstrate that the HI-STORM FW 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].

Chapter 7 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the Confinement Boundary has been rendered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

11.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Chapter 12, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and off-normal handling of HI-TRAC VW) do not result in the degradation of the HI-STORM FW 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.

11.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [11.0.1] specifies the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 11.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 12 presents the results of the evaluations performed to demonstrate that the HI-STORM FW system can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [11.0.1]. The accident events addressed in Chapter 12 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, Confinement Boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, and blockage of MPC basket air inlets.

The worst-case shielding consequence of the accidents evaluated in Chapter 12 for the loaded HI-STORM FW overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded. However, with over 27.5 inches of concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Chapter 12 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. 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 that the resultant dose rate at the 100-meter controlled area boundary would be approximately 2 mrem/hour for the loaded HI-TRAC VW during the accident condition. At the calculated dose rate, it would take approximately 104 days for the dose at the controlled area boundary to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 12. 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.

Table 11.4.1

ANNUAL DOSE FOR ARRAYS OF HI-STORM FW OVERPACKS WITH FUEL LOADING PER TABLE 5.01 (MPC-37)

Array Configuration	1 Cask	1 Cask	2x5 Array
Annual Dose (mrem/year)†	309	13	10
Distance to Controlled Area Boundary (meters)††	100	300	500

† 100% occupancy is assumed.

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

11.5 REFERENCES

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

[11.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989

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

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

[11.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.

[11.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May 1997.

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.

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.

12.1 OFF-NORMAL CONDITIONS

Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions, which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are described in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM FW:

1. Off-Normal Pressure
2. Off-Normal Environmental Temperatures
3. Leakage of One Seal
4. Partial Blockage of Air Inlets
5. Malfunction of FHD System

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of off-normal events and remain in compliance with the applicable acceptance criteria. The following subsections present the evaluation of the HI-STORM FW System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses meet the requirements of 10CFR72.104(a) and 10CFR20, with appropriate margins.

12.1.1 Off-Normal Pressure

The sole pressure boundary in the HI-STORM FW System is the MPC enclosure vessel. The off-normal pressure condition is specified in Subsection 2.2.2. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the MPC cavity under normal storage. The MPC internal pressure under the off-normal condition is evaluated with 10% of the fuel rods ruptured and with 100% of ruptured rods fill gas and 30% of ruptured rods fission gases released to the cavity.

12.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is extremely low. Nonetheless, the event is postulated and evaluated.

12.1.1.2 Detection of Off-Normal Pressure

The HI-STORM FW System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement or safety imperative for detection of off-normal pressure and, therefore, no monitoring is required.

12.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off-normal internal pressure is reported in Subsection 4.6.1 for the following conditions: limiting fuel storage scenario, tech. spec. maximum helium backfill pressure with a 10% rod rupture that causes a 100% of the ruptured rod fill gas and 30% of the ruptured rod gaseous fission products released into the MPC cavity along with off-normal ambient temperature. The analysis shows that the MPC pressure remains below the design MPC internal pressure (given in Table 2.2.1). The corresponding fuel cladding temperature is provided in Table 4.6.1. It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would reduce the temperature rise, therefore the calculated pressure is higher than that would actually occur.

i. Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

ii. Thermal

The MPC internal pressure for off-normal conditions is reported in Subsection 4.6.1. The design basis internal pressure used in the structural evaluation (Table 2.2.1) bounds the off-normal condition 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 features 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, all pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM FW System.

12.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM FW System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. Therefore, there is no corrective action requirement for off-normal pressure.

12.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

12.1.2 Off-Normal Environmental Temperatures

The HI-STORM FW System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (loaded HI-STORM FW overpack) and 0°F to 100°F (loaded HI-TRAC VW transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM FW System and must be evaluated against the allowable component design temperatures. The off-normal temperatures are evaluated against the off-normal condition temperature limits for HI-STORM FW components listed in Table 2.2.3.

12.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM FW System to achieve thermal equilibrium. Because of the large mass of the HI-STORM FW System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

12.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM FW System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM FW overpack and MPC. Chapter 2 provides operational limitations on the use of the HI-TRAC VW transfer cask at temperatures ≤32°F and prohibits use of the HI-TRAC VW transfer cask below 0°F.

12.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considers an environmental temperature of 100°F with insolation for sufficient duration to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The Off-Normal ambient temperature condition is evaluated in Subsection 4.6.1. The results are in compliance with off-normal pressure and temperature limits in Tables 2.2.1 and 2.2.3, respectively.

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a sufficient duration to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM FW overpack. The HI-STORM FW overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC VW transfer cask is 0°F and the HI-TRAC VW is conservatively assumed to reach 0°F throughout the structure. Subsection 3.1.2, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM FW System.

i. Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 4.6.1, the resultant pressure is below the off-normal design pressure (Table 2.2.1). The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits. The effect of the lower off-normal thermal conditions (i.e., -40°F) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsection 3.1.2.

ii. Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Subsection 4.6.1 for the HI-STORM FW overpack and MPC. The evaluation in Subsection 4.6.1 indicates that all temperatures for the off-normal environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.2.3.

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 features 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, all pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM FW System.

12.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM FW System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. For ambient temperatures from 0° to 32°F, ethylene glycol fortified water must be used in the water jacket of the HI-TRAC VW transfer cask to prevent freezing. There are no corrective actions required for off-normal environmental temperatures.

12.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

12.1.3 Leakage of One Seal

The HI-STORM FW System has a high integrity welded boundary to contain radioactive fission products within the Confinement Boundary. The Confinement Boundary is defined by the MPC shell, baseplate, MPC lid, vent and drain port cover plates, closure ring, and associated welds. The closure ring provides a redundant welded closure to further protect against the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. The fabrication shop welds for the confinement boundary are tested for helium leakage. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. The welds on the vent and drain port cover plates are helium leakage tested. Additionally, the MPC lid weld is subjected to a pressure test to verify its integrity. There are no seals present in the design of the MPC confinement boundary.

Section 7.1 provides the narrative that demonstrates that the MPC design, welding, testing and inspection meet the requirements such that leakage from the Confinement Boundary is considered non-credible.

12.1.4 Partial Blockage of Air Inlets

The HI-STORM FW System is designed with debris screens on the inlet and outlet air openings. These screens ensure the openings are protected from the incursion of foreign objects. There are multiple inlet openings and an axisymmetric outlet and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet openings. As required by the design criteria presented in Chapter 2, it is conservatively assumed that 50% of the air inlet openings are completely blocked. The scenario of the partial blockage of air inlets is evaluated with a normal ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the thermal performance of the HI-STORM FW System during this event.

12.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

The presence of screens prevents foreign objects from entering the openings and the screens are either inspected periodically or the outlet air temperature is monitored per the technical specifications. It is, however, possible that blowing debris may partially block the inlet openings for a short time until the openings are cleared of debris.

12.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet openings will occur during the routine visual inspection of the screens or temperature monitoring of the outlet air required by the technical specifications. The frequency of inspection is based on an assumed complete blockage of all air inlet openings. There is no inspection requirement as a result of the postulated partial inlet blockage, because the complete blockage of all air inlet openings is bounding.

12.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

i. Structural

There are no structural consequences as a result of this off-normal event since the HI-STORM FW components do not exceed the off-normal temperature limits (Table 2.2.3).

ii. Thermal

The thermal analysis for the 50% blocked inlet openings off-normal condition is performed in Subsection 4.6.1. The analysis demonstrates that under bounding (steady-state) conditions, no system components exceed the off-normal temperature limits in Table 2.2.3.

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 features 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.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet openings does not affect the safe operation of the HI-STORM FW System.

12.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet openings is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet openings does not affect the safe operation of the HI-STORM FW System.

Periodic inspection of the HI-STORM FW overpack air opening screens is required per the technical specifications. Alternatively, per the technical specifications, the outlet air temperature is monitored. The frequency of inspection is based on an assumed blockage of all air inlet openings analyzed in Section 12.2.

12.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet openings has no radiological impact because the confinement barrier is not breached and the system's shielding effectiveness is not diminished.

12.1.5 Malfunction of FHD System

A FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, a FHD system is equipped with active components requiring external power for normal operation.

12.1.5.1 Postulated Cause of FHD Malfunction

Likely causes of FHD malfunction are (i) a loss of external power to the FHD System and (ii) an active component trips the FHD System. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in helium leakage from the MPC or the FHD.

12.1.5.2 Detection of FHD Malfunction

The FHD System is monitored during its operation. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

12.1.5.3 Analysis of Effects and Consequences of FHD Malfunction

i. Structural

The FHD System is required to be equipped with safety relief devices* to prevent the MPC structural boundary pressures from exceeding the normal condition pressure limits. Consequently there is no adverse effect.

ii. Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit (see Table 2.2.3) must not be exceeded. The FHD System malfunction event is evaluated assuming the following bounding conditions:

* The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.

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.

12.1.5.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

12.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM FW System or events postulated because their consequences may affect the public health and safety. Subsection 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM FW System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. In the following, the evaluation of the design basis postulated accident conditions and natural phenomena is presented. The evaluations demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for postulated accident conditions are defined in Table 2.2.13. The accident load combination evaluations are provided in Section 3.4.

Table 12.2.1 provides a listing of the accident events considered in this section and their probability of occurrence.

12.2.1 HI-TRAC VW Transfer Cask Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-TRAC VW transfer cask is lifted and handled in a vertical orientation at all times. A vertical drop of the loaded HI-TRAC VW transfer cask is not a credible accident as the loaded HI-TRAC VW transfer cask shall be lifted and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-TRAC VW transfer cask in the realm of Part 72 operations is non-credible.

12.2.2 HI-STORM FW Overpack Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-STORM FW overpack is lifted and handled in a vertical orientation at all times. A vertical drop of the loaded HI-STORM FW is not a credible accident as the loaded HI-STORM FW shall be lifted and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-STORM FW in the Part 72 space is non-credible.

12.2.3 HI-STORM FW Overpack Non-Mechanistic Tip-Over

The freestanding HI-STORM FW 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 (Subsection 2.2.3) in Chapter 3.

12.2.3.1 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 Sections 2.2 and 3.4, the methodology and acceptance criteria are presented and a reference tip-over problem is solved. The reference tip-over problem corresponds to a free rotation of the HI-STORM FW 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.

12.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Subsection 3.4.4. 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 impact between the MPC guide tubes and the MPC does not cause a thru-wall penetration of the MPC shell.

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 has been addressed to demonstrate the defense-in-depth features of the design.

12.2.3.3 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.

12.2.4 Fire

12.2.4.1 Cause of Fire

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM FW overpack or HI-TRAC VW transfer cask during their handling.

12.2.4.2 Fire Analysis

The HI-STORM FW System must withstand elevated temperatures due to a fire event. The HI-STORM FW overpack and HI-TRAC VW transfer cask fire accidents for storage are conservatively postulated as described in Subsection 4.6.2. The acceptance criteria for the fire accident are provided in Subsection 2.2.3.

12.2.4.2.1 <u>Fire Analysis for HI-STORM FW Overpack</u>

The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2. 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.

i. Structural

As discussed in Section 3.4, there are no structural consequences as a result of the fire accident condition since the short-term temperature limit on great majority of the concrete is not exceeded and all component temperatures remain within applicable temperature limits (Table 2.2.3). The MPC structural boundary remains within normal condition internal pressure and temperature limits.

ii. Thermal

Based on a conservative analysis discussed in Subsection 4.6.2, of the HI-STORM FW System response to the hypothetical fire event, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM FW System to maintain cooling of the spent nuclear fuel within temperature limits (Table 2.2.3) during and after fire is not compromised.

iii. Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the FSAR." Less than one-inch of the overpack concrete (~4% of the overpack radial concrete thickness) is computed to exceed the short-term temperature limit therefore the effect of this small amount of degraded (not lost) shielding material is not estimated or evaluated in this FSAR.

iv. Criticality

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

v. 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.

vi. Radiation Protection

Since there is minimal reduction, if any, in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM FW System.

12.2.4.2.2 Fire Analysis for HI-TRAC VW Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded HI-TRAC VW transfer cask is performed. The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2.

i. Structural

As discussed in Section 3.4, there are no adverse structural consequences as a result of the fire accident condition.

ii. Thermal

The thermal analysis of the MPC in the HI-TRAC VW transfer cask under a fire accident is performed in Subsection 4.6.2. The analysis shows that the MPC internal pressure and fuel temperature increases as a result of the fire accident. The fire accident MPC internal pressure and peak fuel cladding temperature are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).

As can be concluded from the analysis, the temperatures for fuel cladding and components are below the accident temperature limits.

iii. Shielding

The conservatively assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10CFR72.106 are not exceeded.

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event, since the internal pressure does not exceed the accident condition design pressure and the MPC Confinement Boundary temperatures do not exceed the short-term allowable temperature limits.

vi. Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. Increases in the local dose rates adjacent to the water jacket are evaluated in Chapter 5. Immediately after the fire accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit exposure to site personnel.

12.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC VW transfer cask neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC VW transfer cask in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC VW transfer cask following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The elevated temperatures experienced by the HI-STORM FW overpack concrete shield are limited to the outermost layer of steel and concrete. Therefore, overall reduction in neutron shielding capabilities is quite small. Any increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate. This is due to the limited amount of the concrete shielding which is affected and the already low site boundary dose rates. The loaded HI-STORM FW overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC Confinement Boundary is not compromised and therefore, there is no release of airborne radioactive materials.

12.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC VW transfer cask or HI-STORM FW overpack, the ISFSI owner shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC VW transfer cask as the water jacket rupture discs may open with resulting water loss and increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC VW transfer cask. If damage to the HI-TRAC VW transfer cask is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced. If damage to the HI-TRAC VW transfer cask is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

If damage to the HI-STORM FW storage overpack as the result of a fire event is widespread and/or as radiological conditions require (based on dose rate measurements), the MPC shall be removed from the HI-STORM FW overpack in accordance with Chapter 9. However, the thermal analysis described herein demonstrates that only the outermost layer of the radial concrete which is behind the carbon steel outer shell exceeds its design temperature. The HI-STORM FW overpack may be returned to service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the shielding effectiveness of the overpack is confirmed) and if the visual inspection is satisfactory.

12.2.5 Partial Blockage of MPC Basket Flow Holes

Each MPC basket fuel cell wall has flow holes near the bottom to allow thermosiphon action to assist the cooling of MPC internals. The flow holes in the bottom of the fuel basket in each MPC are located to ensure that the amount of crud listed in Table 2.2.8 does not block the internal helium circulation. Therefore the partial blockage of the HI-STORM FW MPC basket flow holes is not credible.

12.2.6 Tornado

12.2.6.1 Cause of Tornado

The HI-STORM FW System will be stored on an unsheltered ISFSI concrete pad and thus will be subject to ambient environmental conditions throughout the storage period. Additionally, the transfer of the MPC between the HI-TRAC VW transfer cask and the storage overpack may be performed at the unsheltered ISFSI concrete pad. It is therefore possible that the HI-STORM FW System (and/or the HI-TRAC VW transfer cask) may experience the extreme environmental conditions, resulting in the impact from a tornado-borne projectile.

12.2.6.2 Tornado Analysis

A tornado event is characterized by high wind velocities and tornado-generated missiles. The reference missiles considered in this FSAR (see Section 2.2) are of three sizes: small, medium, and large. A small projectile, upon collision with a cask, would tend to penetrate it. A large projectile, such as an automobile, on the other hand, would tend to destabilize a free-standing cask. Accordingly, the tornado event has two distinct effects on the HI-STORM FW System. First, the tornado winds and/or tornado missile attempt to tip-over the loaded HI-STORM FW overpack or HI-TRAC VW transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds, which attempt to penetrate the HI-STORM FW overpack or HI-TRAC VW transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC VW transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC VW transfer cask while it is being handled in the vertical orientation. Penetration by a small missile, however, is credible. The tornado wind and missile are assumed to act synergistically in the safety evaluation in Section 3.4 to determine the kinematic stability of the HI-STORM FW overpack.

i. Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the HI-STORM FW overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or large tornado missiles.

Analyses provided in Section 3.4 also show that there is a potential for a tornado missile (8 inch steel cylinder) to penetrate the water jacket of the HI-TRAC VW transfer cask. The HI-STORM FW overpack will suffer minor local damages due to the missile impact with no significant damage in the shielding and there will be no damage to the MPC.

ii. Thermal

The thermal consequences of the complete loss of water due to rupture of the water jacket from a tornado missile has been analyzed in Section 4.6. It has been demonstrated that the consequences are within the short term fuel cladding and material temperature limits.

iii. Shielding

Since the structural evaluation shows that the tornado missiles may penetrate the HI-TRAC VW water jacket and cause loss of water, for a conservative estimate of the dose rates a complete loss of water in the water jacket is assumed and is bounded by the fire condition assumptions. This assumption results in an increase in the radiation dose rates however the shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded.

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

HI-STORM FW overpack: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Since there is only a possibility of minimal reduction in localized shielding and there is 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.

HI-TRAC VW transfer cask: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Increases in the local dose rates as a result of the possible loss of water in the HI-TRAC VW transfer cask water jacket are evaluated in Chapter 5. Immediately after the tornado accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit the exposure to the site personnel.

12.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded HI-STORM FW overpack; damage the shielding materials of the HI-STORM FW overpack or HI-TRAC VW transfer cask; or damage the MPC Confinement Boundary. There is no affect on the radiation dose as a result of the tornado winds.

A tornado missile may cause localized damage in the concrete radial shielding of the HI-STORM FW overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC VW transfer cask water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC

VW transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

12.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM FW System to a tornado, the ISFSI owner shall perform a visual and radiological inspection of the overpack and/or HI-TRAC VW transfer cask.

Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and may be repaired, if required, while in-service. The HI-STORM FW overpack may continue its service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the final visual inspection is satisfactory.

Damage sustained by the HI-TRAC VW transfer cask shall be inspected and repaired. As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. If damage to the HI-TRAC VW transfer cask water jacket or HI-TRAC VW transfer cask body is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

12.2.7 Flood

12.2.7.1 Cause of Flood

Many ISFSIs are located in flood plains susceptible to floods. Therefore, it is necessary for such ISFSIs to define a Design Basis Flood (DBF). The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

A flood event is characterized by two parameters:

a. flood water velocity
b. flood height over the ISFSI pad as a function of time

The design basis flood (DBF) event for an ISFSI site should provide the maximum flood water velocity. The highest flood height, on the other hand, is not the governing condition for the flood event because of the vented construction of the overpack. The bottom vents in the HI-STORM FW overpack ensure that the flood level inside and outside the overpack will be equal. When the flood waters are high and the MPC is fully submerged then there is no short-term threat to the storage system because the MPC's heat rejection to water is far more efficient than the (normal condition) heat rejection to air. The most adverse flood condition, therefore, exists when the flood waters are high enough to block the inlet openings but no higher. In this scenario, the MPC surface has minimum submergence in water and the ventilation air is completely blocked. In fact, as the flood water begins to accumulate on the ISFSI pad, the air passage size in the inlet vents begins to get smaller. Therefore, the rate of floodwater rise with time is necessary to analyze the thermal-hydraulic problem. For the reference design basis flood (DBF) analysis in this FSAR, the flood waters are assumed to rise instantaneously to the height to block the inlet vents and stay at that precise elevation for 32 hours. For each ISFSI site subject to a DBF, the flood time-history will be incorporated in determining the acceptability of the flood event. The acceptance criteria are provided in Section 2.2.

The analysis results for the reference DBF are presented below.

12.2.7.2 Flood Analysis

i. Structural

The flood accident affects the HI-STORM FW overpack structure in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood accident affects the MPC by applying an external pressure.

Section 3.4 provides the analysis of the flowing floodwater applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

ii. Thermal

As stated above, for a flood of sufficient height to allow the water to come into extensive contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a submerged flood will enter the annulus between the MPC and the overpack. The water would provide cooling that would be an order of magnitude greater than that available in the air filled annulus (due to water's higher heat transfer coefficient).

The reference DBF that is most adverse to heat rejection is treated for 100% blockage of air inlets in Subsection 12.2.13. If the duration of the flood blockage exceeds the DBF blockage specified in Subsection 4.6.2 then a site specific evaluation shall be performed in accordance with the methodology presented in Chapter 4 and evaluated for compliance with Subsection 2.2.3 criteria.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water provides additional shielding that reduces radiation doses.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the fuel pool, which is presented in Section 6.1.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring Confinement Boundary integrity.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM FW System.

12.2.7.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

12.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM FW System sustains no damage as a result of the flood, which is a short-duration event. At the completion of the flood, exposed surfaces may need debris and adherent foreign matter removal.

12.2.8 Earthquake

12.2.8.1 Cause of Earthquake

The HI-STORM FW System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM FW System, the ISFSI may experience an earthquake. The earthquake event postulated for the ISFSI is referred to as the "Design Basis Earthquake" (DBE). The DBE is defined in Subsection 2.2.3.

12.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM FW overpack based on a static stability criteria discussed in Subsection 2.2.3.

Some ISFSI sites will have earthquakes that exceed the static stability limit specified in Subsection 2.2.3. For these high-seismic sites, a dynamic analysis shall be performed based on the methodology provided in Subsection 3.4.4.

i. Structural

The methodology for the evaluation of the earthquake consequences has been presented in Subsection 3.4.4. An earthquake is a vibratory event, which is fully described by an acceleration time-history. However, for "weak" earthquakes, a static equilibrium based calculation suffices. However, for stronger DBEs, a dynamic analysis is required. The results from the reference DBE analyzed in Subsection 3.4.4 are used to evaluate the safety case for the earthquake using the acceptance criteria in Section 2.2.

The following conclusions can be reached from the structural analysis of the HI-STORM FW System under the reference DBE event:

a. The MPC Confinement Boundary remains unbreached.
b. The HI-STORM FW overpack structure remains intact; i.e., the lid is not displaced.
c. There is no physical damage to the HI-STORM FW overpack shielding concrete.
d. The HI-STORM FW overpack does not tip over.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

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

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the reference DBE does not affect the continued safe operation of the HI-STORM FW System.

12.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates at the site boundary.

12.2.8.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have displaced from their installed position or tipped over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 12.2.3.

12.2.9 100% Fuel Rod Rupture

This accident event postulates the non-mechanistic condition that all the fuel rods rupture and that the quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity consistent with ISG-5, Revision 1.

12.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM FW System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. Therefore, there is no credible cause for 100% fuel rod rupture. This accident is presumably postulated in NUREG-1536 to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

12.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no containment consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, the shielding capability of the system, or the criticality control features of the fuel basket; and does not challenge the structural integrity of the MPC.

i. Structural

The structural analysis provided in Chapter 3 evaluates the MPC Confinement Boundary under the accident condition design internal pressure limit set in Table 2.2.1. Calculations in Chapter 4 show that the accident internal pressure limit bounds the pressure from 100% fuel rod rupture.

ii. Thermal

The determination of the maximum accident pressure is provided in Chapter 4. The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.5, which is bounded by the design basis accident condition MPC internal pressure limit set in Table 2.2.1.

iii. Shielding

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

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident event does not affect the safe operation of the HI-STORM FW System.

12.2.9.3 100% Fuel Rod Rupture Dose Calculations

The breach of fuel cladding postulated in this accident event does not result in any physical change to the storage system other than some release of gases and a limited quantity of solids (particulates) into the gaseous helium space. The amount of the radiation source remains unaffected. Hence, the radiation dose at the site boundary will not change perceptibly, i.e., there are no consequences to the site boundary dose.

12.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC Confinement Boundary is not damaged. The HI-STORM FW System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

12.2.10 Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified would cause failure of the confinement boundary. The MPC uses redundant confinement closures to assure that there is no release of radioactive materials. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The information contained in Chapter 7 demonstrates that MPC is designed, fabricated, tested and inspected to meet the guidance of ISG-18 such that unacceptable leakage from the Confinement Boundary is non-credible.

12.2.11 Explosion

12.2.11.1 Cause of Explosion

An explosion within the protected area of an ISFSI is improbable since there are no explosive materials permitted within the site boundary. However, an explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity the effects of an explosion on a reinforced structure are minimal. Explosions that are credible for a specific ISFSI would require a site hazards evaluation under the provisions of 72.212 regulations by the ISFSI owner using the methodology set forth in Section 3.1.

12.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external design pressure (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 12.2.7 and the tornado missile accident of Subsection 12.2.6, because explosive materials are not stored within close proximity to the casks. The bounding analysis shows that the MPC and the overpack can withstand the effects of substantial accident external pressures without collapse or rupture.

An ISFSI where a credible explosion event produces a pressure wave greater than analyzed in Subsection 12.2.7 or an impactive load greater than considered in Subsection 12.2.6, shall be evaluated within the purview of §72.212. The results of the safety evaluation of the postulated limiting explosion in this subsection are as follows:

i. Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

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

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring Confinement Boundary integrity.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the reference explosion accident does not affect the safe operation of the HI-STORM FW System.

12.2.11.3 Explosion Dose Calculations

The reference bounding external pressure load has no effect on the HI-STORM FW Overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM FW System is experienced as a result of the explosion pressure load. The effects of explosion generated (reference) missiles on the HI-STORM FW System structure is bounded by the analysis of tornado generated missiles.

12.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by an explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM FW

System. Following an explosion, the ISFSI owner shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the overpack will be repaired as necessary.

12.2.12 Lightning

12.2.12.1 Cause of Lightning

As the HI-STORM FW System will be stored on an unsheltered ISFSI concrete pad, there is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

12.2.12.2 Lightning Analysis

The HI-STORM FW System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM FW overpack is struck with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, provides a direct path to the ground through the grounding cable.

The MPC provides the Confinement Boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

i. Structural

There is no structural consequence as a result of this event.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

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

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STORM FW System.

12.2.12.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the Confinement Boundary or shielding materials. Therefore, no further analysis is necessary.

12.2.12.4 Lightning Accident Corrective Action

The HI-STORM FW System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

12.2.13 100% Blockage of Air Inlets

12.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all bottom inlets. A blockage of all of the circumferentially arrayed inlets cannot be realistically postulated to occur at most sites. However, a flood, blizzard snow accumulation, tornado debris, or volcanic activity, where applicable, can cause a significant blockage.

12.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet openings 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 opening, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the HI-STORM FW overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass and correspondingly large thermal capacity of the HI-STORM FW overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for an extended duration. This accident condition is, however, a short duration event that will be identified by the ISFSI staff, at worst, during scheduled periodic surveillance at the ISFSI site and corrected using the site's emergency response process.

i. Structural

There are no structural consequences as a result of this event.

ii. Thermal

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

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. Radiation Protection

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

Based on the above evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the blockage is the first priority activity.

12.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets accident, the shielding capabilities of the HI-STORM FW overpack are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

12.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within the maximum elapsed period between scheduled surveillance inspections. Upon detection of the complete blockage of the air inlet openings, the ISFSI owner shall activate its emergency response procedure to remove the blockage with mechanical and manual means as necessary. After clearing the overpack openings, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for the assurance that the temperature limits are not exceeded.

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

12.2.14 Burial Under Debris

12.2.14.1 Cause of Burial Under Debris

Complete burial of the HI-STORM FW System under debris is not a credible accident. During storage at the ISFSI, there are no structures above the casks that may collapse and surround them. The minimum regulatory distance(s) from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM FW System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM FW overpack air inlet openings has already been considered in Subsection 12.2.12.

12.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM FW System does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. A perverse effect of the overlaid debris would be to provide additional shielding to reduce radiation doses. The accident external pressure considered in this FSAR during the flood bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STORM FW System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limit during a burial under debris accident, assuming that the debris has the consistency of a typical pile of rocks (pebbles).

i. Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions set in Table 2.1.1 bounds the pressure calculated for this event. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

ii. Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised even if the burial event lasts for a substantial duration.

iii. Shielding

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

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring Confinement Boundary integrity.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no deleterious effect on the site boundary dose a result of this event.

Based on the above evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the debris is the first priority activity.

12.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM FW System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no adverse radiological impact.

12.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures are not exceeded even for an extended duration of burial. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC VW transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the implementation of this corrective action.

12.2.15 Extreme Environmental Temperature

12.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated (see Table 2.2.2) as a 3-day average temperature caused by extreme weather conditions.

12.2.15.2 Extreme Environmental Temperature Analysis

To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration (3 days) to allow the HI-STORM FW overpack to achieve thermal equilibrium.

The accident condition considering an extreme environmental temperature (Table 2.2.2) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3.

i. Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

ii. Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in evaluation performed in Subsection 4.6.2. As concluded from this evaluation, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

iv. Criticality

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

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring Confinement Boundary integrity.

vi. Radiation Protection

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

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM FW System.

12.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM FW System for the extreme environmental temperature and the dose calculations are to the same as those for normal condition dose rates.

12.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

Table 12.2.1

ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	Event	Probability of Occurrence	Subsection Where Addressed	Comments
1.	HI-TRAC VW Transfer Cask Handling Accident	Non-Credible	12.2.1	This FSAR mandates the use of single-failure-proof lifting devices for handling loaded HI-TRAC VWs within the Part 72 jurisdictional boundary.
2.	HI-STORM FW Overpack Handling Accident	Non-Credible	12.2.2	This FSAR mandates the use of single-failure-proof lifting devices for handling loaded HI-STORM FWs within the Part 72 jurisdictional boundary.
3.	HI-STORM FW Non-Mechanistic Tip-Over	Non-Credible	12.2.3	The HI-STORM FW tip-over event is more properly referred to as a "non-mechanistic" tip-over, meaning that no physical loading considered in this FSAR leads to a tip-over event.
4.	Fire	Very small probability but credible	12.2.4	Although there are no ignition sources in the ISFSI area, combustible material (motive fuel) is present. Therefore, the potential of a fire event cannot be ruled out categorically.
5.	Partial Blockage of MPC Basket Vent Holes	Non-Credible	12.2.5	An impactive event may jolt the stored fuel and cause its crud to fall off. However, as explained in Subsection 12.2.5, there is no realistic mechanism for the blockage of the flow holes.
6.	Tornado	Credible	12.2.6	Because a HI-STORM FW ISFSI can be deployed in any state within the U.S., the potential of a tornado event at a generic ISFSI must be considered.
7.	Flood	Credible	12.2.7	Flood, like tornado, must be categorized as a credible event at a generic ISFSI site.
8.	Earthquake	Credible	12.2.8	The Design Basis Earthquake for an ISFSI is a specified event for a nuclear facility.

Table 12.2.1

ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	Event	Probability of Occurrence	Subsection Where Addressed	Comments
9.	100% Fuel Rod Rupture	Non-Credible	12.2.9	"100% rod rupture" is a non-mechanistic event; no specific loading event has been identified to cause 100% rod rupture.
10.	Confinement Boundary Leakage	Non-Credible	12.2.10	The Confinement Boundary has been determined to be invulnerable to leakage in Chapter 7.
11.	Explosion	Credible	12.2.11	Explosion of gasoline is a credible event at an ISFSI.
12.	Lightning	Credible	12.2.12	Lightning is a small probability event at any ISFSI, hence, it cannot be deemed non-credible.
13.	100% Blockage of Air Inlets	Non-Credible	12.2.13	Because the air openings are along the circumference of the cask, and surveillance is at very short intervals (see Technical Specification), the assumption of blockage of all openings has no mechanistic basis.
14.	Burial Under Debris	Credible	12.2.14	Burial of a loaded system under a debris cannot be generically ruled out because a nuclear plant site may (ever so minimally) susceptible to a large adverse environment event such as a tsunami or an avalanche.
15.	Extreme Environmental Temperature	Credible	12.2.15	In certain desert areas in the country a temperature spike that reaches the accident temperature limit (Table 2.2.2) cannot be ruled out. Such areas have not been declared unfit at a nuclear plant site by the USNRC and therefore, must be factored in defining generic accident events.

12.3 OTHER EVENTS

This section addresses the MPC reflood event which is placed in the category of "other events" since it cannot be categorized as an off-normal or accident event. The MPC reflood event occurs if an ISFSI owner needs to return the fuel in a loaded canister to wet storage in the plant's fuel pool. The MPC reflood event is presented in Subsection 9.4.3 in connection with the preparation for the MPC unloading operation. The reflooding of a loaded MPC with water results in a change to the environment around the fuel from a gaseous (low heat transfer medium) to aqueous (high heat transfer medium). This implies the generation of thermal stresses in the fuel cladding and potential for loss of cladding integrity.

The safety analysis of the reflood event in Subsection 3.4.4 focuses on the effect of strains (due to reflooding) on the integrity of the cladding. This safety analysis, which uses an appropriate thermal/structural model as well as evaluations in Subsection 4.5.5, forms the basis for the instructions in the MPC reflood procedures provided in Subsection 9.4.3. For a complete evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel, the postulated cause of the event, monitoring of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented in this section similar to the systematic evaluations presented for design basis off-normal and postulated accident condition events in the preceding sections.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of MPC reflood and remain in compliance with the applicable acceptance criteria. In particular, the integrity of the fuel cladding shall be preserved. The following subsections contain the evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel that demonstrates that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses continue to meet the requirements of 10CFR72.104(a) and 10CFR20.

12.3.1 MPC Reflood

MPC reflood is performed during the preparation for the unloading operations as described in Subsection 9.4.3. The MPC is flooded with water at a controlled rate as specified in Subsection 4.5.5 with the MPC vent port open such that the generation of steam from flashing of water is not excessive and the pressure within the MPC remains below its normal condition internal design pressure. Although past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by a direct introduction of water into the canister space (as specified in Subsection 9.4.3) a structural evaluation has been performed to ensure fuel cladding integrity and is provided in Paragraph 3.4.4.1.

12.3.1.1 Postulated Cause of MPC Reflood

Likely causes to perform MPC reflood include those associated with required actions for certain limiting conditions for operation (LCO) (as specified in Appendix A of the Technical Specifications) to unload fuel assemblies from the MPC. A reflood operation may also be carried out at the plant owner's volition to return the fuel to wet storage as a voluntary act.

12.3.1.2 Monitoring of MPC Reflood

MPC reflooding is monitored at frequent intervals by the surveillance of MPC pressure as required by SR 3.1.3.1 to Specification 3.1.3 (Appendix A of the Technical Specifications). An indication of MPC pressurization at or above normal condition MPC internal design pressure established in Section 2.2 requires the immediate action to stop reflooding operations until the MPC cavity pressure is below the required limit. See LCO 3.1.3 and the associated basis in Chapter 13 for more information.

12.3.1.3 Analysis of Effects and Consequences of MPC Reflood

i. Structural

MPC Enclosure Vessel Integrity: The MPC water reflood rate specified in Subsection 4.5.5, the essential reflooding control procedure steps established in Subsection 9.4.3, and the surveillance instructions in SR 3.1.3.1 Specification 3.1.3 (Appendix A of the Technical Specifications) ensure that the MPC is maintained below the normal condition pressure limit and well below MPC off-normal and accident condition pressure limits set down in Section 2.2, thus ensuring large margins of safety and no harmful effect on the MPC enclosure vessel integrity.

Fuel Cladding Integrity: The structural evaluation in Paragraph 3.4.4.1 ensures that the fuel cladding integrity is preserved during the reflood event.

Other Structural Related Considerations: MPC reflooding is performed under a specified maximum water injection rate and below normal condition MPC internal design pressure. The pressures and temperatures are therefore compatible with design limits of existing MPC ancillaries and standard connections such as RVOAs. Maintaining the pressure and temperature parameters well below the design basis values for the ancillaries ensures that failure of components and appurtenances during the reflooding operation is unlikely. Therefore, no credible mechanism for risk to the plant staff or general public from radiological release due to the reflood operation can be identified.

ii. Thermal

A thermal evaluation is provided in Subsection 4.5.5 to specify the maximum water reflood rate. The maximum calculated water reflood rate will prevent MPC over-pressurization and fuel cladding damage.

iii. Shielding

There is no adverse effect on the shielding performance of the system as a result of the MPC reflood event. The shielding performance of the MPC is indeed enhanced by the flooding of its contents.

iv. Criticality

There is no adverse effect on the criticality control of the system as a result of this planned plant event. The essential procedure steps in Chapter 9 and surveillance SR 3.3.1.1 to Specification 3.3.1 (Appendix A of the Technical Specifications) ensure that the water used to reflood the MPC will have the minimum required soluble boron concentration. The generation and escape of steam from the MPC will increase (not lower) the soluble boron concentration.

v. Confinement

During the reflood operation, the MPC confinement function is inoperative (and supplanted by the part 50 facility) as the canister is connected to the plant's fluid accumulation system and the source of water (such as the fuel pool). The reflooding operation, however, does not degrade the confinement capability of the MPC because the internal pressures and temperatures are procedurally controlled to remain well within the design limits.

vi. Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no adverse effect on occupational or public exposures as a result of this MPC reflood event. The vent port steam is delivered to the radwaste gas facility of the plant in accordance with the specified procedure in Subsection 9.4.3.

Based on this evaluation, it is concluded that MPC reflood has no adverse effects or consequences on the safety or operability of the HI-STORM FW System.

12.3.1.4 Corrective Action for MPC Reflood

See Specification 3.1.3 (Appendix A of the Technical Specifications) on MPC Cavity Reflooding and Specification 3.3.1 (Appendix A of the Technical Specifications) on Boron Concentration.

12.3.1.5 Radiological Impact of MPC Reflood

The event has no radiological impact because the plant's confinement barrier and shielding infrastructure are unaffected and the operation relies on no new system for the control of effluents.

12.4 REFERENCES

Currently no references listed.

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 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.

Table 13.1.1

HI-STORM FW SYSTEM CONTROLS

Condition to be Controlled	**Applicable Technical Specifications**†
Criticality Control	3.3.1 Boron Concentration
Confinement boundary integrity and integrity of cladding on undamaged fuel	3.1.1 Multi-Purpose Canister (MPC)
Shielding and radiological protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 MPC Reflooding 3.2.1 TRANSFER CASK Surface Contamination 5.1 Radioactive Effluent Control Program 5.3 Radiation Protection Program
Heat removal capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural integrity	5.2 Transport Evaluation Program

† Technical Specifications are located in Appendix A to the CoC. Authorized contents are specified in this FSAR in Subsection 2.1.8

Table 13.1.2 HI-STORM FW SYSTEM TECHNICAL SPECIFICATIONS	
NUMBER	**TECHNICAL SPECIFICATION**
1.0	USE AND APPLICATION
1.1	DEFINITIONS
1.2	LOGICAL CONNECTORS
1.3	COMPLETION TIMES
1.4	FREQUENCY
2.0	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SFSC Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Surface Contamination
3.3	SFSC Criticality Control
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINSTRATIVE CONTROLS
5.1	Radioactive Effluent Control Program
5.2	Transport Evaluation Program
5.3	Radiation Protection Program

13.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM FW system to assure long-term performance consistent with the conditions analyzed in this FSAR.

13.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM FW Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM FW System Design (overview);

2. ISFSI Facility Design (overview);

3. Systems, Structures, and Components Important-to-Safety (overview);

4. HI-STORM FW System Safety Analysis Report (overview);

5. NRC Safety Evaluation Report (overview);

6. Certificate of Compliance conditions;

7. HI-STORM FW Technical Specifications, Approved Contents, Design Features and other Conditions for Use;

8. HI-STORM FW Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);

9. Required instrumentation and use;

10. Operating Experience Reviews

11. HI-STORM FW System and ISFSI Procedures, including
    - Procedural overview
    - Fuel qualification and loading
    - MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
    - MPC welding operations
    - HI-TRAC/overpack staging operation
    - Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, supplemental cooling (if used), and cooldown)

- MPC/HI-TRAC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

13.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM FW system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. Dry run training already performed successfully for the HI-STORM 100 System can be substituted for dry run steps applicable to HI-STORM FW. The dry run shall include, but is not limited to the following:

1. Receipt inspection of the HI-STORM FW System components.

2. Moving the MPC/HI-TRAC into the spent fuel pool.

3. Preparation of the MPC/HI-TRAC for fuel loading.

4. Selection and verification of specific fuel assemblies to ensure conformance.

5. Locating specific assemblies and placing assemblies into the MPC/HI-TRAC (using a dummy fuel assembly), including appropriate independent verification.

6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.

7. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup MPC may be used).

8 Placement of the HI-STORM FW System at the ISFSI.

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.

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 2.1.9 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.

13.2.8 MPC

a. Basket material composition, properties, dimensions, and tolerances for criticality control.

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

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

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

13.2.9 HI-STORM FW Overpack

a. HI-STORM overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during handling and storage operations.

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

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

13.2.10 HI-TRAC VW Transfer Cask

a. HI-TRAC transfer cask material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations.

b. HI-TRAC transfer cask material thermal properties and dimensions for heat transfer control.

c. HI-TRAC transfer cask material composition and dimensions for dose rate control.

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

The examples below execute the methodology and equations described in Subsection 2.1.9 for determining allowable decay heat, burnup, and cooling time for the approved contents.

Example 1

In this example, it will be assumed that the MPC-37 is being loaded with array/class 17x17A fuel in its regionalized loading pattern as shown in Figure 1.2.1 with heat loads from Table 1.2.3.

Table 13.2.1 provides four hypothetical fuel assemblies in the 17x17A array/class that will be evaluated for acceptability for loading in the MPC-37. The decay heat values and the fuel classification in Table 13.2.1 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for storage in Region 2 of MPC-37. Fuel Assembly Number 1 is not acceptable for storage in Region 1 or Region 3 because the total heat load of the fuel assembly and the non-fuel hardware exceeds the decay heat limit for those regions.

Fuel Assembly Number 2 is not acceptable for loading. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-37 (Figure 2.1.1). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-37.

Fuel Assembly Number 3 is acceptable for loading in Region 1 or Region 2. The fuel assembly is limited to these locations due to the non-fuel hardware (Figure 2.1.5) and the total heat load of the fuel assembly and non-fuel hardware is less than the decay heat limits for these regions.

Fuel Assembly Number 4 is not acceptable for loading in the MPC-37 because its cooling time is less than the minimum of 3 years. When the fuel assembly attains three years cooling time it can be reevaluated based on the decay heat.

Example 2

In this example, it will be assumed that the MPC-89 is being loaded with array/class 10x10A fuel in its regionalized storage pattern as shown in Figure 1.2.2 with heat loads from Table 1.2.4.

Table 13.2.2 provides four hypothetical fuel assemblies in the 10x10A array/class that will be evaluated for acceptability for loading in the MPC-89. The decay heat values and the fuel classification in Table 13.2.2 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for loading in the MPC-89. Fuel Assembly 1 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit higher than the decay heat of the assembly, therefore the assembly is acceptable for loading in the MPC-89, but it is limited to the cells depicted in Figure 2.1.2

Fuel Assembly Number 2 is not acceptable for loading in the MPC-89. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-89.

Fuel Assembly Number 3 is acceptable for loading in Regions 1, 2 or 3 of the MPC-89.

Fuel Assembly Number 4 is acceptable for loading in Region 2 of the MPC-89 only. The fuel assembly is limited to these locations due to the total heat load of the fuel assembly.

Table 13.2.1

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 17x17A)

FUEL ASSEMBLY NUMBER	1	2	3	4
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	2.9
FUEL ASSEMBLY DECAY HEAT (KW)	1.01	1.45	0.4	2.08
NON-FUEL HARDWARE STORED WITH ASSEMBLY	BPRA	None	NSA	None
NFH DECAY HEAT (KW)	0.5	0	0.3	0
FUEL CLASSIFICATION	Undamaged	Damaged	Undamaged	Undamaged

Table 13.2.2

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 10x10A)

FUEL ASSEMBLY NUMBER	**1**	**2**	**3**	**4**
INITIAL ENRICHMENT (WT. % ^{235}U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	7
FUEL ASSEMBLY DECAY HEAT (KW)	0.43	0.55	0.2	0.61
FUEL CLASSIFICATION	Damaged	Damaged	Undamaged	Undamaged

13.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM FW system are provided in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC. Bases applicable to the Technical Specifications are provided in the FSAR Appendix 13.A. The format and content of the HI-STORM FW system Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, "Writer's Guide for the Restructured Technical Specifications" [13.3.1] was used as a guide in the development of the Technical Specifications and Bases.

13.4 REGULATORY EVALUATION

Table 13.1.2 lists the Technical Specifications for the HI-STORM FW system. The Technical Specifications are detailed in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC.

The conditions for use of the HI-STORM FW system identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical Specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM FW system will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

13.5 REFERENCES

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

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

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

[13.3.1] Nuclear Management and Resources Council, Inc. – *Writer's Guide for the Restructured Technical Specifications,* NUMARC 93-03, February 1993.

HI-STORM FW SYSTEM FSAR

APPENDIX 13.A

TECHNICAL SPECIFICATION BASES

FOR THE HOLTEC HI-STORM FW MPC STORAGE SYSTEM

BASES TABLE OF CONTENTS

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES	
LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).
LCO 3.0.2	LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

(continued)

BASES	
LCO 3.0.2 (continued)	Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications. The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.
LCO 3.0.3	This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 3.0.4	LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM FW System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist: a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to exit the Applicability desired to be entered to comply with the Required Actions. Compliance with Required Actions that permit continuing with dry fuel storage activities for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the dry storage system. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

(continued)

BASES	
LCO 3.0.4 (continued)	The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC. Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing to demonstrate: a. The equipment being returned to service meets the LCO; or b. Other equipment meets the applicable LCOs. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES	
SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
SR 3.0.1	SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO. Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when: a. The systems or components are known to not meet the LCO, although still meeting the SRs; or b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances. Surveillances do not have to be performed when the HI-STORM FW System is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary dry storage cask system parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow dry fuel storage activities to proceed to a specified condition where other necessary post maintenance tests can be completed.

(continued)

BASES

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.
The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

(continued)

BASES

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of HI-STORM FW System conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

(continued)

BASES	
SR 3.0.3 (continued)	If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance. Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.
SR 3.0.4	SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability. This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe conduct of dry fuel storage activities. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability. However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

(continued)

BASES	
SR 3.0.4 (continued)	The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC. The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 SFSC Integrity

B 3.1.1 Multi-Purpose Canister (MPC)

BASES	
BACKGROUND	A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then, the MPC vent and drain port cover plates and closure ring are installed and welded. Inspections are performed on the welds.
	MPC cavity moisture removal using vacuum drying or forced helium dehydration is performed to remove residual moisture from the MPC cavity space after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel.
	If forced helium dehydration is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.
	After the completion of drying, the MPC cavity is backfilled with helium meeting the requirements of the CoC.
	Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Backfilling the MPC with helium in the required quantity eliminates air in-leakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage.

(continued)

BASES	
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the confinement boundary of the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC over-pressurization evaluation.
LCO	A dry, helium filled, and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the fuel cladding. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.
APPLICABILITY	The dry, sealed, and inert atmosphere is required to be in place prior to TRANSPORT OPERATIONS to ensure both the confinement and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively, during these periods in support of other activities being performed with the stored fuel.

(continued)

BASES

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the cavity vacuum drying pressure or demoisturizer exit gas temperature limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS (continued)

B.1
If the helium backfill quantity limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

B.2
Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition either by adding or removing helium or by demonstrating through analysis that all system limits will continue to be met. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

C.1
If the helium leak rate limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM FW OVERPACK is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES	
ACTIONS (continued)	C.2 Once the consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION. D.1 If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.
SURVEILLANCE REQUIREMENTS	SR 3.1.1.1 , SR 3.1.1.2, and SR 3.1.1.3 The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Under certain conditions, cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the gas temperature or dew point at the specified location reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. Other conditions require the forced helium dehydration method of moisture removal to be used to provide necessary cooling of the fuel during drying operations.

(continued)

BASES	
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. 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

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES	
BACKGROUND	The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system that ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the inlet air ducts. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the outlet air ducts at the top of the OVERPACK.
APPLICABLE SAFETY ANALYSIS	The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the inlet and outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs. Analyses have been performed for the complete obstruction of half, and all inlet air ducts. Blockage of half of the inlet air ducts reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits. The complete blockage of all inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the duration of the analyzed event.

(continued)

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.
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)

BASES	
ACTIONS (continued)	A.1 Although the heat removal system remains operable, the blockage should be cleared expeditiously. B.1 If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path. C.1 If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the system user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures. C.2.1 In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented. This Required Action must be complete in 24 hours. The Completion Time is consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 32 hours after event initiation.

(continued)

BASES

ACTIONS (continued)

C.2.1 (continued)

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

C.2.2

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

An engineering evaluation must be performed to determine if any concrete deterioration has occurred in the OVERPACK which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Time of 24 hours reflects the Completion Time from Required Action C.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads.

(continued)

BASES	
SURVEILLANCE REQUIREMENTS	SR 3.1.2 The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE. Visual observation that all inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Greater than 50% blockage of the total inlet or outlet air duct area renders the heat removal system inoperable and this LCO not met. 50% or less blockage of the total inlet or outlet air duct area does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s). As an alternative, for an OVERPACK with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limit stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility. The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.
REFERENCES	1. FSAR Chapter 4 2. ANSI/ANS 57.9-1992

B 3.1 SFSC INTEGRITY

B 3.1.3 MPC Cavity Reflooding

BASES	
BACKGROUND	In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The pressure in the MPC cavity is ensured to be less than the 100 psig design pressure. This is accomplished via direct measurement of the MPC gas pressure or via analysis. After ensuring the MPC cavity pressure meets the LCO limit, the MPC is then reflooded with water at a controlled rate and/or the pressure monitored to ensure that the pressure remains below 100 psig. Once the cavity is filled with water, the MPC lid weld is removed leaving the MPC lid in place. The TRANSFER CASK and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and TRANSFER CASK are removed from the spent fuel pool and decontaminated. Ensuring that the MPC cavity pressure is less than the LCO limit ensures that any steam produced within the cavity is safely vented to an appropriate location and eliminates the risk of high MPC pressure due to sudden generation of large steam quantities during re-flooding.
APPLICABLE SAFETY ANALYSIS	The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the MPC in which the fuel assemblies are stored. Standard practice in the dry storage industry has historically been to directly reflood the storage canister with water. This standard practice is known not to induce fuel cladding failures. The integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by introducing water to the cavity in a controlled manner such that there is no sudden formation of large quantities of steam during MPC reflooding. (Ref. 1).

(continued)

BASES	
LCO	Determining the MPC cavity pressure prior to and during re-flooding ensures that there will be sufficient venting of any steam produced to avoid excessive MPC pressurization.
APPLICABILITY	The MPC cavity pressure is controlled during UNLOADING OPERATIONS after the TRANSFER CASK and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the MPC Reflood LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. A note has been added to the APPLICABILITY for LCO 3.1.3 which states that the LCO is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concerning for dry UNLOADING OPERATIONS.
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. A If the MPC cavity pressure limit is not met, actions must be taken to restore the parameters to within the limits before initiating or continuing re-flooding the MPC. Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.

(continued)

BASES	
SURVEILLANCE REQUIREMENTS	SR 3.1.3.1 The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal pressure prior to and during re-flooding the MPC, sufficient steam venting capacity exists during MPC re-flooding. The LCO must be met on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved.
REFERENCES	1. FSAR Chapters 3, 4, 9 and 12.

B 3.2 SFSC Radiation Protection

B 3.2.1 TRANSFER CASK Surface Contamination

BASES	
BACKGROUND	A TRANSFER CASK is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the TRANSFER CASK may become contaminated with the radioactive material from the spent fuel pool water. This contamination is removed prior to moving the TRANSFER CASK to the ISFSI, or prior to transferring the MPC into the OVERPACK, whichever occurs first, in order to minimize the radioactive contamination to personnel or the environment. This allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.
APPLICABLE SAFETY ANALYSIS	The radiation protection measures, implemented during MPC transfer and transportation using the TRANSFER CASK, are based on the assumption that the exterior surfaces of the TRANSFER CASK have been decontaminated. Failure to decontaminate the surfaces of the TRANSFER CASK could lead to higher-than-projected occupational doses.
LCO	Removable surface contamination on the TRANSFER CASK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm^2 from beta and gamma sources and 20 dpm/100 cm^2 from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the TRANSFER CASK loading process.

(continued)

BASES	
LCO (continued)	Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose. LCO 3.2.1 requires removable contamination to be within the specified limits for the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC. The location and number of surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the MPC means the upper portion of the MPC external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed. The user shall determine a reasonable number and location of swipes for the accessible portion of the MPC. The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC, while implementing sound ALARA practices. A Note for this LCO indicates that the limits on surface contamination are not prescribed for the TRANSFER CASK if MPC TRANSFER is to occur inside the FUEL BUILDING, however plant radiation protection procedures should guide these limits for occupational dose considerations.
APPLICABILITY	The applicability is modified by a note that states that the LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING. This is consistent with the intent of this LCO, which is to ensure loose contamination on the loaded TRANSFER CASK and MPC outside the FUEL BUILDING is within limits. If the MPC transfer is performed inside the FUEL BUILDING the empty TRANSFER CASK remains behind and is treated like any other contaminated hardware under the user's Part 50 contamination control program. Verification that the surface contamination is less than the limit in the LCO is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS, when the LCO is applicable. Measurement of surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.

(continued)

BASES	
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. A subsequent use of the TRANSFER CASK that does not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1</u> If the removable surface contamination of a TRANSFER CASK or MPC, as applicable, which has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the TRANSFER CASK or MPC and bring the removable surface contamination to within limits. The Completion Time of 7 days is appropriate given that sufficient time is needed to prepare for, and complete the decontamination once the LCO is determined not to be met.
SURVEILLANCE REQUIREMENTS	SR 3.2.1.1 This SR verifies that the removable surface contamination on the TRANSFER CASK and/or accessible portions of the MPC is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification during LOADING OPERATIONS in order to confirm that the TRANSFER CASK or OVERPACK can be moved to the ISFSI without spreading loose contamination.
REFERENCES	1. FSAR Chapter 9 2. NRC IE Circular 81-07.

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, 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.a provides the minimum concentration of soluble boron required in the MPC water for the MPC-37. 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 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 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)

BASES

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. If the boron concentration of water in the MPC is less than its limit, all LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions must be suspended immediately.

A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately. One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration is initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours prior to entering the Applicability of the LCO. For LOADING OPERATIONS, this means within four hours of loading the first fuel assembly into the MPC using two independent measurements to ensure the requirements of 10 CFR 72.124(a) are met. These two independent measurements will be repeated every 48 hours while the MPC is submerged in water or if water is to be added to or recirculated through the MPC.

For UNLOADING OPERATIONS, this means verifying the boron concentration in the source of borated water to be used to reflood the MPC within four hours of commencing reflooding operations and every 48 hours after until all the fuel is removed from the MPC. Two independent measurements will be taken to ensure the requirements of 10 CFR 72.124(a) are met. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premise of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed. After the completion of the surveillance methods, events which might change the soluble boron concentration will be administratively controlled per the LCO. If actions are taken that could result in a reduction in the boron concentration the surveillance will be performed again.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure

BASES	
SURVEILLANCE REQUIREMENTS	that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO.
REFERENCES	1. FSAR Chapter 6.

CHAPTER 14[†]: QUALITY ASSURANCE PROGRAM

14.0 INTRODUCTION

14.0.1 Overview

This chapter provides a summary of the quality assurance program implemented by Holtec International for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the Company's dry storage/transport systems including the HI-STORM FW system which includes the HI-TRAC VW transfer cask. This chapter is included in this FSAR to fulfill the requirements in 10 CFR 72.140 (c) (2) and 72.2(a)(1),(b).

Important-to-safety activities related to construction and deployment of the HI-STORM FW system are controlled under the NRC-approved Holtec Quality Assurance Program. The Holtec QA program manual [14.0.1] is approved by the NRC [14.0.2] under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.

1. Holtec Quality Assurance Program Manual
2. Holtec Quality Assurance Procedures
3. a. Holtec Standard Procedures

   b. Holtec Project Procedures

Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide the assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61.

14.0.2 Graded Approach to Quality Assurance

Holtec International uses a graded approach to quality assurance on all safety-related or important-to-safety projects. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Subsection 14.0.1.

NUREG/CR-6407 [14.0.3] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STORM FW system and HI-TRAC VW transfer cask. The ITS categories assigned to the HI-STORM FW cask components are identified in Tables 2.0.1 through 2.0.8. Quality categories for ancillary equipment are provided in Chapter 9 of this FSAR. Quality categories for other equipment needed to deploy the HI-STORM FW system at a licensee's ISFSI are defined on a case-specific basis considering the component's design function using the guidelines of NUREG/CR-6407 [14.0.3].

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM FW system at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. These activities include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM FW structures, systems, and components (SSCs) that are important-to-safety.

The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [14.0.4]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536 and has accordingly been added to the list of deviations in Table 1.0.3. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities.

14.1 REFERENCES

[14.0.1] Holtec International Quality Assurance Program, Latest Approved Revision.

[14.0.2] NRC QA Program Approval for Radioactive Material Packages No. 0784, Docket 71-0784.

[14.0.3] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[14.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.