1.2.1.5 Neutron and Gamma Shielding Features

The HI-STAR 190 Cask Containment is circumscribed by the Gamma Capture Space (GCS) and the Neutron Capture Space (NCS), described in the foregoing, that respectively attenuate gamma radiation and neutron fluence emitted from the contained fuel to minimal practical levels consistent with ALARA principles. The HI-STAR 190 Packaging (with or without the personnel barrier) ensures the external radiation standards of 10 CFR 71.47 under exclusive shipment are met when loaded with design basis fuel. The drawing package in Section 1.3 and the summary description in Section 1.1 provide information on the configuration of neutron and gamma shielding features.

While most of the shielding in the transport package is contained in the body of the cask and specifically in the Gamma Capture Space (GCS) and the Neutron Capture Space (NCS) described in Section 1.1, a certain amount of shielding is also provided by the Fuel Basket, the Basket Shims, and the Enclosure Vessel. The arrangement of the shielding materials shown in the licensing drawings reflects the shielding optimization carried out for the HI-STAR 190 cask.

During transport, the impact limiters provide additional gamma shielding (steel) at the ends of the cask and help prevent loss of shielding as a result of normal and accident conditions of transport by encapsulation of the containment top forging and the bottom forging regions. Note that for normal conditions of transport, the impact limiters are not credited for shielding, except for the stand-off distance they provide from the cask body.

~~Critical~~ Physical Characteristics of the Holtite Neutron Shielding Material used in the safety analyses are provided in Table 1.5.3.

1.2.1.6 Criticality Control Features

Criticality control in the HI-STAR 190 Packaging is provided by the coplanar grid work of the Fuel Basket honeycomb, made entirely of the Metamic™-HT extruded borated metal matrix composite plates. Thus the entire body of the Fuel Basket, made exclusively from Metamic-HT, serves as the neutron absorber in the HI-STAR 190 Packaging. Therefore, unlike baskets made of steel, the neutron absorber is not attached to the cell walls by mechanical means that may be vulnerable to detachment. Hence, the locational fixity of the neutron absorber is guaranteed.

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Metamic-HT was first certified by the USNRC in 2009 for use in the HI-STAR 180 transport application under Docket No. 71-9325 as the sole constituent material for the fuel basket types F-37 and F-32 for transporting high burn up and MOX fuel. Subsequently, MPC-68M, a Metamic-HT equipped fuel basket for BWR fuel was certified in the HI-STORM 100, Docket No. 72-1014. All

fuel baskets presently used in HI-STORM FW (Docket No. 72-1032), HI-STORM UMAX (Docket No. 72-1040) and HI-STAR 180D (Docket No. 71-9367) utilize Metamic-HT for neutron absorbing and structural functions.

Additionally, for transporting PWR fuel, burnup credit is applied as an additional criticality control feature, following the guidance in ISG-8 Rev. 3 [1.2.12]. Since the Metamic basket material contains the B-10 neutron absorber at a significant higher level than steel based baskets, the burnup requirements for PWR fuel are comparatively low.

Finally, there are no moderators in the HI-STAR 190 Packaging.

1.2.1.7 Lifting and Tie-Down Devices

Lifting trunnions are attached ~~to~~ via the trunnion support structure to the neutron shield ribs~~cask containment closure flange~~ for lifting and also for rotating the cask body between vertical and horizontal positions. Two lifting trunnions are located 180° apart in the sides of the top flange. Two additional trunnions are attached near the bottom extremity of the cask and located 180° apart. ~~The bottom trunnions may be slightly off-center to facilitate the rotation direction of the cask.~~

The pair of top lifting trunnions is conservatively qualified to independently lift the cask in compliance with ANSI N14.6 increased stress margins as specified in Section 8.1.

Lifting trunnions are designed in accordance with 10CFR71.45 and ANSI N14.6 [1.2.5], manufactured from a high strength alloy and installed in threaded openings. These trunnions are designed to meet the requirements of 10 CFR 71, as detailed in Chapter 2. The trunnions are also designed to collapse in the event of a drop accident to a completely flush position. Upending and down-ending are typically performed with the cask pivoting on an ancillary tilting device specifically designed for this purpose.

The last image in Figure 1.2.5 provides an illustration of a typical package rail transport configuration. The support saddles provide attachment points for belly slings/straps around the cask body to prevent excessive vertical or lateral movement of the cask during normal transportation. The impact limiters affixed to both ends of the cask are designed to transmit the design basis axial transport loads into the longitudinal stops.

1.2.1.8 Heat Transfer Features

The HI-STAR 190 Package can safely transport SNF by maintaining the fuel cladding temperature below the limits for normal and accident conditions consistent with the guidance in the NRC Interim Staff Guidance, ISG-11 Rev. 3 [1.2.6]. The temperature of the fuel cladding is dependent on the decay heat and the heat dissipation capabilities of the cask. The SNF decay heat is passively dissipated without any mechanical or forced cooling. The primary heat transfer mechanisms in the HI-STAR 190 Package are conduction and thermal radiation.

Table 1.5.3: ~~Critical~~ Physical Characteristics of Holtite-B

Property (Note 1)	Property Value
Bulk Density, g/cm^3	See Table 8.1.9
Hydrogen Areal Density, g/cm^2	See Table 8.1.9
Nominal Boron Carbide Content, wt%	~~See Table 8.1.9~~2
Design Temperature, °C (°F)	204 (400)

Notes:
1~~.~~ Properties with characteristics needed for shielding are defined in Table 8.1.9
2. All properties are critical characteristics except for Boron Carbide Content

APPENDIX 1.A: PROPRIETARY APPENDIX WITHHELD PER 10 CFR 2.390

Table 2.1.3: Stress Limits for Lid Closure Bolts (Elastic Analysis per NB-3230)

Stress Category	Level A	Level D
Average Service Stress	$2S_m$	Cannot exceed Yield Strength
Maximum Service Stress (tension + bending but no stress concentrations)	$3S_m$	Joint Remains Leak Tight (see Note 2). Cannot exceed Ultimate Strength

Notes:

1. Stress limits for Level A loading ensure that bolt remains elastic.
2. Limit set on primary tension plus primary bending for Level D loading is based on an elastic stress evaluation; however, the overriding acceptability of the joint design is performance based on an assured absence of leakage.
3. Since the cask closure lid bolt joint is a friction type connection as explained below, the bolts are exempt from shear stress evaluation per ASME Section III, F-1135.2 and NF-3324.6. The closure lid bolt joint is sufficiently preloaded (3.035×10^6 lbf per Table 2.2.10), and the friction force (based on a conservatively assumed static friction coefficient of 0.5) at the lid-to-flange interface can prevent the 13,000 lb lid from sliding as long as the lid's deceleration is less than 117 g's. Per Table 2.7.2 of the SAR, the maximum lid deceleration in all analyzed 30 ft drop events is 89.0 g's. Therefore, the bolt joint is qualified as a friction type connection per ASME Section III, NF-3324.1.
   ~~2.~~

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2.2.1.2 Nonstructural Materials

2.2.1.2.1 Gamma Shielding Material

Lead is not considered as a structural member of the HI-STAR 190 Package. However, it is included in the dynamic simulation models for Normal and Accident Conditions of Transport. Applicable mechanical properties of lead are provided in Table 2.2.9.

2.2.1.2.2 Neutron Shielding Material

The non-structural properties of the neutron shielding material Holtite B are provided in Section 1.2. Holtite B does not serve a structural function in the HI-STAR 190 package.

2.2.1.2.3 Fuel Basket Supports

Representative mechanical properties for the basket supports are tabulated in Table 2.2.7. Table 2.2.4 provides the mechanical properties for the stainless steel basket support.

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2.2.1.2.4 Cask Coating

The HI-STAR 190 cask's exterior~~internal~~ surfaces are coated with a conventional surface preservative such as Carboguard® 890 (see www.carboline.com for product data sheet) and/or

Table 2.6.8: PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

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

[2.6.5] Holtec Proprietary Report HI-2084137, "A Classical Dynamics Based and Experimentally Benchmarked Impact Response Computation Methodology for AL-STAR Equipped Casks", Latest Revision.

[2.6.6] NUREG/CR-1132, "A Survey of Potential Light Water Reactor Fuel Rod Failure Mechanisms and Damage Limits", Courtright, E.L., July 1979.

[2.6.7] Lanning, D. D. and Beyer, C. E., "Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rods During Short Term Operations for Dry Cask Storage", Pacific Northwest National Laboratory, January 2004.

[2.7.1] Holtec Proprietary Report HI-981891, "Impact Limiter Test Report - Second Series", Rev 3, 1998.

[2.7.2] Holtec Proprietary Report HI-2063591, "Benchmarking of LS-DYNA for Simulation of Hypothetical Drop Conditions of Transport", Rev. 1, 2007.

[2.7.3] Holtec Proprietary Report HI-2094418, "Structural Calculation Package for HI-STORM FW System", Revision 11.

[2.7.4] Holtec Proprietary Report HI-2073743, "Benchmarking the LS-DYNA Impact Response Prediction Model for the HI-STAR Transport Package Using the AL-STAR Impact Limiter Test Data", Rev. 1, 2008.

[2.7.5] Shah, M.J., Klymyshyn N.A., and Kreppel B.J., "HI-STAR 100 Spent Fuel Transport Cask Analytic Evaluation for Drop Events", Packaging, Transport, and Security of Radioactive Materials, Vol. 18, No. 1, W.S. Maney & Sons (2007).

[2.7.6] Containment Performance of Transportable Storage Casks at 9-m Drop Test, Hitoshi Tobita and Kenji Araki, PATRAM 2004, Berlin, Germany, 9/2004.

[2.7.7] "Mechanical Design of Heat Exchangers and Pressure Vessel Components", by K.P. Singh and A. I. Soler, Arcturus Publishers, Cherry Hill, New Jersey, 1100 pages, hardbound (1984).

[2.7.8] Holtec Proprietary Report HI-2073715, Benchmarking of LS-DYNA For Use With Polyurethane Foam Filled Impact Limiter, 2007, Revision 0.

[2.7.9] "Validation of an Impact Limiter Crush Prediction Model with Test Data: The Case of the HI-STAR 100 Package", K.P. Singh, A.I. Soler, and C. Bullard, PATRAM 2004, Berlin, Germany, September 20-24, 2004.

CHAPTER 5 - SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STAR 190 Package to demonstrate compliance with 10CFR71.47 and 10CFR71.51 is presented in this chapter. HI-STAR 190 is designed to accommodate either MPC-37 or MPC-89, containing up to 37 PWR and 89 BWR fuel assemblies, respectively.

In order to offer the user flexibility in fuel loading, the HI-STAR 190 Package offers several different loading patterns, where different positions in the basket are qualified for different burnup/cooling time/enrichment combinations. The loading patterns used for shielding evaluations are described in Appendix 7.C. All loading patterns were analyzed and found to be acceptable compared to the regulatory limits.

In addition to storing intact PWR and BWR fuel assemblies, the HI-STAR 190 system is designed to transport BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Glossary. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

The transport index in 10CFR71 is defined as the number determined by multiplying the radiation level in milliSievert per hour (mSv/h) at one meter from the external surface of the package by 100. Since HI-STAR 190 is designed to meet the dose rate limit of 10 mrem/hr (0.1 mSv/h) at 2 meters from the surface of the vehicle, the dose rate at 1 meter from the package could be greater than 10 mrem/hr (0.1 mSv/h) and the transport index could exceed 10. Therefore, HI-STAR 190 loaded with design basis fuel must be shipped by exclusive use shipment as discussed in Chapter 1.

The shielding analyses were performed with MCNP-5 1.51 [5.0.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H [5.0.2] and ORIGEN-S [5.0.3] sequences from the SCALE 5.1 systems. These are principally the same codes that were used in Holtec's approved Storage and Transportation FSARs and SAR under separate docket numbers [5.0.4]. Detailed descriptions of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

This chapter contains the following information:

- A description of the shielding features of HI-STAR 190.
- A description of the source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for HI-STAR 190.
- Analyses for the HI-STAR 190's content and results to show that the 10CFR71.47 dose rate limits are met during normal conditions of transport and that the 10CFR71.51 dose rate limit is not exceeded following hypothetical accident conditions.

SAS2H/ORIGEN-S calculations related to reactor input parameters, decay heat generation, and source term calculations.

5.2.1 Gamma Source

Table 5.2.3(a) and Table 5.2.3(b) provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for selected burnup and cooling time combinations utilized in the shielding calculations of the assemblies in MPC-37 and MPC-89.

NUREG-1617 [5.2.2] states that "In general, only gammas from approximately 0.8 MeV-2.5 MeV will contribute significantly to the external radiation levels." [

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ORIGEN-S was used 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 from ^{59}Co, steel and inconel was 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 5.2.4. These scaling factors were taken from Reference [5.2.3].

Table 5.2.5(a) and Table 5.2.5(b) provide the ^{60}Co activity utilized in the shielding calculations

Burnable poison rod assembly (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. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis WE 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 5.2.4 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.

The following assumptions are applied for BPRAs and TPDs source term.

1. The non-fuel hardware devices are conservatively assumed to be present in every fuel assembly.
2. Cooling time of BPRAs and TPDs is the same as the cooling time of the corresponding fuel assembly.
3. The burnup of BPRA is assumed to be 40000 MWD/MTU.
4. TPDs source terms are based on a maximum TPD Co-60 activity. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the cobalt-60 production rate as the initial cobalt-59 impurity is being depleted.
4.5. A ^{59}Co impurity level of 1 g/kg is used for the steel components while 4.7 g/kg for Inconel components.

TABLE 5.2.12

DESCRIPTION OF AXIAL POWER SHAPING ROD

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of Cladding (kg ~~Inconel~~Steel)	Mass of Absorber (kg ~~AgInCd~~Inconel)
Start (in.)	Finish (in.)	Length (in.)			
Configuration 1 - 10% Inserted					
0.0	15.0	15.0	1.0	1.26	5.93
15.0	18.8125	3.8125	0.2	0.32	1.51
18.8125	28.25	9.4375	0.1	0.79	3.73

TABLE 5.2.13

DESIGN BASIS SOURCE TERMS FOR AXIAL POWER SHAPING ROD

Axial Dimensions Relative to Bottom of Active Fuel			Curies Co-60 ~~from Inconel~~
Start (in.)	**Finish (in.)**	**Length (in.)**	
Configuration 1 - 10% Inserted			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78

Table 5.4.7 provides the dose rates at various locations on the surface and two meter from HI-STAR 190 due to the BPRAs and TPDs for MPC-37. Table 5.4.8 provides the dose rates at various locations at a distance of two meters from HI-STAR 190 due to the BPRAs, TPDs, CRAs and APSRs for MPC-37 loaded with Westinghouse 17x17 fuel assembly.

The results in Table 5.4.7 and Table 5.4.8 show that the maximum dose effect for BPRAs is at the side of the cask, the maximum dose effect for TPDs is at the top of the cask and the maximum dose effect for APSRs is at the bottom of the cask. All dose rates with NFH in this chapter are based on APSRs in Regions 1 and 2 and BPRAs in Region 3 fuel assemblies.

5.4.9 SONGS Maximum Dose Rates

The San Onofre Nuclear Generating Station (SONGS) utilized PWR CE16x16 fuel assemblies. CE16x16 fuel is bounded by W17x17 and B&W15x15 fuel types. However, SONGS site specific shielding evaluations are performed (to allow loading of fuel assemblies with shorter cooling times) as discussed in this subsection.

The loading patterns used for SONGS shielding evaluations are described in Appendix 7.C. All loading patterns were analyzed and found to be acceptable compared to the regulatory limits.

The maximum SONGS dose rates are calculated with non-fuel hardware (with the bounding APSR). The typical CE16x16 non-fuel hardware is control rod assemblies (CRAs) and axial power shaping rods (APSRs). The permissible locations of the CRAs and APSRs are provided in Appendix 7.C.

Dose rates were calculated on the cask surface and at a distance of 2 m from the package surfaces, on locations shown in Figure 5.1.1. Results are presented in Tables 5.4.9 and 5.4.10.

Figure 5.1.2 shows the dose locations at 1 meter from the surface for the conditions of the HI-STAR 190 Package after the postulated accident. Corresponding maximum dose rates are listed in Table 5.4.11.

All values in these tables are below the regulatory limits.

5.4.10 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, shielding evaluations were previously performed for the 100-ton HI-TRAC and HI-STAR 100 [5.0.4] 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 cask.

As a defence in depth, additional sensitivity analyses for fuel reconfiguration in normal and accident conditions are provided in Subsection 5.4.5, demonstrating that all analyzed regionalized loading patterns meet the dose rate regulatory requirements.

Calculations documented in Chapter 2 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 damage to the neutron absorber on the outside of the cask. However, this external absorber is already neglected in the calculational models. Other parameters important to criticality safety are fuel type, fuel burnup 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. There are, however, differences between the normal and accident models in terms of internal and external water density, external condition and level of potential fuel reconfiguration. The effect of these conditions is discussed in Subsections 6.3.4 and 6.3.5.

6.3.2 Material Properties

Composition of the various components of the principal designs of the HI-STAR 190 package is listed in Table 6.3.5. The nuclide identification number (ZAID), presented for each nuclide in Table 6.3.5, includes the atomic number, mass number and the cross-section evaluation identifier, which are consistent with the ZAIDs used in the benchmarking calculations documented in Appendix A. In this table only the composition of fresh fuel is listed. For a discussion of the composition of spent fuel for burnup credit see Appendices 6.B and 6.C.

HI-STAR 190 is designed such that the fixed neutron absorber will remain effective for a period greater than 50 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of the fixed neutron absorber are provided in Paragraph 1.2.1.6 and Chapter 2.

As specified in Table 8.1.4, the manufacturer's minimum B_4C content for the Metamic-HT fixed neutron absorber is 10 wt%. The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Paragraph 8.1.5.5, to validate the ^{10}B (poison) concentration in the fixed neutron absorber. In addition, based on calculations performed in the HI-STORM FW FSAR [6.0.4], the fraction of ^{10}B atoms destroyed during the service life in the fixed neutron absorber by neutron absorption is negligible (less than 10^{-7}). Therefore, there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

The only materials affected by the accident conditions are the Holtite neutron absorber on the outside of the cask, and the impact limiters. None of these materials are considered in the criticality model. Therefore, material properties of the materials used in the criticality analyses are not affected by the accidents.

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The calculations demonstrate that the thick wall of the overpack is more than sufficient to preclude neutron coupling between casks, consistent with the findings of Cano [6.3.4], et al. Neglecting the Holtite neutron shielding in the calculational model provides further assurance of conservatism in the calculations.

6.3.4.2 Partial Flooding

To demonstrate that HI-STAR 190 would remain subcritical if water were to leak into the containment system, as required by 10CFR71.55, calculations in this section address partial flooding in HI-STAR 190 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 the MPC-37 and MPC-89 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 partial density (0.0002 g/cm^3). Results of these calculations are shown in Table 6.3.12(a). In general, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. The fully flooded case therefore represents the bounding condition for all MPC basket types.

6.3.4.3 Pellet-to-Clad Gap Flooding

The reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated in Subsection 6.2.2. The results, presented in Table 6.2.1, confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. Thus, for all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded.

6.3.4.4 Preferential Flooding

Preferential flooding of the MPC basket is not possible [

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6.3.4.5 Eccentric Positioning of Assemblies in Fuel Storage Cells

TABLE 6.3.12(a)

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TABLE 6.3.12(b)

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

Appendix 6.B

PROPRIETARY APPENDIX WITHHELD PER 10 CFR 2.390

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the mating device open end during MPC lowering due to radiation streaming. The mating device may be used to supplement shielding during removal of the MPC lift rigging.

11. Lower the MPC into HI-STAR 190.

12. Disconnect the MPC lifting slings from the lifting device.

13. Remove HI-TRAC VW from on top of HI-STAR 190 with or without the HI-TRAC bottom lid.

14. Remove the MPC lift rigging and install plugs in the empty MPC bolt holes*.

15. Remove the mating device from on top of HI-STAR 190.

7.1.4 Cask Closure

1. The test port plugs on the inter-seal test ports of the closure lid and closure lid port covers are installed with new seals and torqued. The containment closure flange's sealing surface protective cover is removed. The sealing surfaces on the closure lid are inspected for signs of damage or particulate matter that might affect the seal performance. Any particulate matter or sealing surface damage that would prevent a seal is remedied. The closure lid is installed using either new or existing seals. The closure lid bolts are installed and torqued. Bolt torque requirements and recommended tightening procedure are provided in Table 7.1.1 and Figure 7.1.1, respectively. The user may attach security seals to the outer closure lid bolts at this time.
2. If the MPC contains HBF, then the cask cavity is leak tested to the required acceptance criteria in Chapter 8. Unacceptable leakage rates will require unloading of the MPC. Note – this leak test is for MPCs that contain HBF. The leak test of the overpack, which is performed regardless of MPC contents, is described below.
3. The cask cavity is dried, evacuated and backfilled to the requirements in Table 7.1.2.
4. The closure lid access port plug, fitted with a new seal, is closed.
5. The closure lid inner-seal and closure lid access port plug seal are leak tested to the required acceptance criteria in Chapter 8. Unacceptable leakage rates will require cleaning or repair of the sealing surfaces and replacement of the seals prior to retesting of the seals.

7.1.5 Preparation for Transport

1. A periodic leakage test of the overpack's containment boundary shall be performed

* Upon installation, studs, nuts, and threaded plugs shall be cleaned and inspected for damage or excessive thread wear (replaced if necessary) and coated with a light layer of Loctite N-5000 High Purity Anti-Seize (or equivalent).

INDEX TO APPENDIX 7.C

Table 7.C.1 (Page 1 of 2)

FUEL ASSEMBLY LIMITS

I. MPC MODEL: MPC-37

A. Allowable Contents

1. Uranium oxide, PWR undamaged fuel assemblies, damaged fuel assemblies, and/or fuel debris, meeting the criteria in Table 7.C.2, with or without non-fuel hardware and meeting the following specifications (Note 1):

a.	Fuel Rod Cladding Material, Guide Tubes Material and Instrument Tubes Material ~~type~~:	ZR
b.	Maximum initial enrichment:	See Table 7.C.4
c.	Post-irradiation cooling time, average burnup per assembly:	Cooling time ≥ 3 years Assembly average Burnup ≤ 68.2 GWD/MTU
d.	Decay heat per assembly	See Table 7.C.7
e.	Fuel assembly length:	≤ 199.2 inches (nominal design including non-fuel hardware and DFC)
f.	Fuel assembly width:	≤ 8.54 inches (nominal design)
g.	Fuel assembly weight:	≤ 2050 lbs (including ~~non-fuel hardware~~NFH and DFC)
h.	Maximum Initial Uranium Loading:	≤ 0.495 MTU

B. Quantity per MPC: Up to 37 fuel assemblies can be stored in MPC-37 in one of the configuration listed in Table 7.C.5.

C. One (1) Neutron Source Assembly (NSA) is authorized for loading in MPC-37

D. Up to thirty (30) BPRAs are authorized for loading in MPC-37

E. Minimum cooling time of NFH is equal to minimum cooling time required of the fuel assembly containing NFH per Table 7.C.7 and Table 7.C.8. ~~Example is provided in table 7.C.8(a) , Note 1.~~In addition, NFH must satisfy specification provided in Table 7.C.13. An example is provided in notes to Table 7.C.8(a).

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts, with or without ITTRs, may be stored in any fuel storage location. Fuel assemblies containing APSRs, RCCAs, CEAs, CRAs, or NSAs may only be loaded in fuel storage Regions 1 and 2 (see Figure 7.C.1)

Table 7.C.1 (Page 2 of 2)

FUEL ASSEMBLY LIMITS

II. MPC MODEL: MPC-89

A. Allowable Contents

1. Uranium oxide, BWR undamaged fuel assemblies, damaged fuel assemblies, and/or fuel debris meeting the criteria in Table 7.C.3, with or without channels and meeting the following specifications (Note 1):

a. Fuel Rod Cladding Material, Water Rod Material and Channel Material~~Cladding type~~:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table 7.C.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	5 wt. % of U-235
d. Post-irradiation cooling time, average burnup,:	
i. Array/Class 8x8F:	Cooling time ≥ 10 years and an assembly average burnup ≤ 27.5 GWD/MTU
ii. All other Array Classes:	Cooling time ≥ 3 years and an assembly average burnup ≤ 65 GWD/MTU
e. Decay heat per assembly:	
i. Array/Class 8x8F:	≤183.5 Watts
ii. All Other Array Classes:	See Table 7.C.9
f. Fuel assembly length:	≤ 176.5 inches (nominal design)
g. Fuel assembly width:	≤ 5.95 inches (nominal design)
h. Fuel assembly weight:	≤ 850 lbs, including a DFC as well as a channel
i. Maximum Initial Uranium Loading:	≤ 0.198 MTU

B. Quantity per MPC: Up to 89 fuel assemblies can be stored in MPC-89 in one of the configuration listed in Table 7.C.6.

Note 1: The lowest maximum allowable enrichment of any fuel assembly loaded in an MPC-89, based on fuel array class and fuel classification, is the maximum allowable enrichment for the remainder of the assemblies loaded in that MPC.

Note 1

Example: Qualifying a fuel assembly in class 17x17A with burnup of 44,000 MWD/MTU , enrichment 3.1 wt % and cooling time of 5 years and heat load of 1160 W and inserted NFH (BPRAs) with cooling time of 4 years and 20,000MWD/MTU burnup and heat load of 17 W into Region 2 of Loading Pattern 1 in Table 7.C.7:

1. **Allowable Heat Load:** In Table 7.C.7 find the maximum allowable decay heat load per basket cell for the loaded region and the loading pattern. In this example the maximum decay heat load per basket cell in region 2 of loading pattern 1 is 1.7 kW which is greater than the heat load of the fuel assembly 1160 W combined with 17 W heat load from inserts.
2. **Burnup:** In Table 7.C.8(a) locate value of burnup that is equal or greater than 44,000 MWD/MTU. In this example the closest higher value is 45,000 MWD/MTU.
3. **Enrichment:** In Table 7.C.8(a) confirm that the minimum enrichment next to selected maximum burnup is lower or equal to fuel assembly enrichment. In this example the minimum enrichment next to 45,000 MWD/MTU is 3 wt% which is lower than assembly enrichment of 3.1 wt%.
4. **Cooling Time:**
   Confirm that fuel assembly cooling time is equal or greater than value in Table 7.C.8(a). In this example, the minimum cooling time is found in the intersection of row for assembly with burnup of 45,000 MWD/MTU and enrichment of 3 wt% and column for minimum decay heat of 1.7 kW. The minimum cooling time is thus 3.5 years which is in this example lower than the actual assembly cooling of 5 years.
   Confirm that NFH cooling time is equal or greater than value in Table 7.C.8(a) and Table 7.C.13. In this example, the NFH is inserted in a fuel assembly for which the minimum required cooling time is 3.5 years. The minimum required cooling time for NFH with 20,000 MWD/MTU -burnup in Table 7.C.13 is ~~thus also 3.54~~ years ~~which is lower than the actual NFH cooling time of 4 years~~.
5. **Conclusion:** Assembly with burnup of 44,000 MWD/MTU, enrichment 3.1 wt % and cooling time of 5 years and inserted NFH with burnup of 20,000 MWD/MTU and cooling time of 4 years is acceptable for loading into Region 2 of Loading Pattern 1.

Note 2: Fuel specification in Table 7.C.8(a) is applicable to all fuel assembly classes in Table 7.C.2.

Note 3: Cooling times are limited to 60 years. Dash line (-) means that the fuel combination cannot be loaded in the corresponding region.

Table 7.C.13

NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, 3, and 8)

Post-irradiation Cooling Time (yrs)	Inserts (Note 4) Maximum Burnup (MWD/MTU)	NSA or Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)	Control Component (Note 6) Maximum Burnup (MWD/MTU)	APSR Maximum Burnup (MWD/MTU)
≥ 3	≤ 24,635	N/A (Note 7)	N/A	N/A
≥ 4	≤ 30,000	≤ 20,000	N/A	N/A
≥ 5	≤ 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ 60,000	≤ 45,000	-	≤ 83,000
≥ 9	-	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 90,000	-	-
≥ 13	-	≤ 180,000	-	-
≥ 14	-	≤ 630,000	-	-

NOTES:

1. Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.
2. Linear interpolation between points is permitted, except that NSA or Guide Tube Hardware and APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 14 years and ≥ 11 years, respectively.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts.
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
7. N/A means not authorized for loading at this cooling time.
8. Non-fuel hardware burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post-irradiation.

APPENDIX 7.D

BURNUP VERIFICATION CONDITIONS OF THE HI-STAR 190 PACKAGE

For those spent fuel assemblies that need to meet the burnup requirements specified in Table 7.C.4(a)~~Table 7.D.6~~, a burnup verification shall be performed in accordance with either Method A or Method B described below.

Method A: Burnup Verification Through Quantitative Burnup Measurement

For each assembly in the F-37 where burnup credit is required, the minimum burnup is determined from the burnup requirement applicable to the configuration chosen for the cask (see Table 7.C.4(a)~~Table 7.D.6~~). A measurement is then performed that confirms that the fuel assembly burnup exceeds this minimum burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. The assembly burnup value to be compared with the minimum required burnup should be the measured burnup value as adjusted by reducing the value by a combination of the uncertainties in the calibration method and the measurement itself.

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

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

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

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

8.2 MAINTENANCE PROGRAM

An ongoing maintenance program for the HI-STAR 190 Cask and impact limiter will be prepared and issued prior to the delivery and first use of the HI-STAR 190 Package as a part of its O&M Manual. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued radiological safety, proper handling, and containment performance of the HI-STAR 190 Package in accordance with 10CFR71, Ref [8.0.1], regulations, conditions in the Certificate of Compliance, and the design requirements and criteria contained in this Safety Analysis Report (SAR). An MPC maintenance program under the aegis of the HI-STORM FW FSAR (Docket # 72-1032) or the HI-STORM UMAX FSAR (Docket # 72-1040) shall apply with specific requirements provided in this Section.

The HI-STAR 190 package is totally passive by design. There are no active components or systems required to assure the continued performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from weathering effects, and pre- and post-usage requirements for transportation. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces, seal replacement, and leak testing following seal replacement. Such maintenance requires methods and procedures no more demanding than those currently in use at nuclear power plants.

A maintenance inspections and tests program schedule for the HI-STAR 190 Package is provided in Table 8.2.1.

8.2.1 Structural and Pressure Tests

No periodic structural or pressure tests on the packaging following the initial acceptance tests are required to verify continuing performance.

The MPC maintenance program under the aegis of the HI-STORM FW FSAR (Docket # 72-1032) or the HI-STORM UMAX FSAR (Docket # 72-1040) shall include an aging management program (applicable to storage durations longer than the initial storage license life) that ensures continued radiological safety of the MPC and verifies that the MPC pressure and/or containment boundary (as applicable) is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce the effectiveness of the packaging. See Appendix 8.A.

8.2.2 Leakage Tests

A pre-shipment leakage rate test of cask containment seals and MPC containment boundary (See Appendix 8.A for applicability) is performed per Subsection 8.1.4 following ~~fuel~~ loading ~~per Subsection 8.1.4~~of the sealed MPC in the HI-STAR 190 Overpack. This pre-shipment leakage rate test is valid for 1 year. If the pre-shipment leakage rate test expires, a periodic leakage rate test of the containment seals must be performed prior to transport. This periodic leakage rate test is valid for 1 year.