Designated Original

71-9261

HOLTEC INTERNATIONAL

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July 27, 2005

ATTN: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference: 1. USNRC Docket No. 71-9261 (HI-STAR 100), TAC No. L23796
2. Holtec Project 5014
3. SFPO Letter dated 23 April 2004

Subject: Request for Additional Information for HI-STAR 100 Transportation Package

Dear Sir:

By application dated 16 September 2003 and as supplemented by letter dated 23 March 2004, we requested approval of an amendment to the Certificate of Compliance for our HI-STAR 100 Transportation Package (Reference 1). Via letter on 23 April 2004 (Reference 3), the SFPO issued a Request for Additional Information (RAI) on this amendment. We herein submit our responses to the 23 April 2004 RAI.

The following non-proprietary attachments are provided:

Non-Proprietary Attachment 1: Written Responses to Non-Proprietary RAIs (9 pages)
Non-Proprietary Attachment 2: Updated Non-Proprietary Proposed Revised SAR Pages (342 pages)
Non-Proprietary Attachment 3: Updated Marked-Up Certificate of Compliance (50 pages)
Non-Proprietary Attachment 4: Updated Revised Certificate of Compliance (50 pages)
Non-Proprietary Attachment 5: Affidavit Pursuant to 10 CFR 2.390 (5 pages)

The following proprietary attachments are provided:

Proprietary Attachment 1: Written Responses to Proprietary RAIs (13 pages)
Proprietary Attachment 2: Updated Proprietary Proposed Revised SAR Pages (177 pages)

We note that the NRC has withheld the information presented in both of these proprietary attachments from public disclosure in the past.

Document ID: 5014573

NMSS01

HI-STAR 100 CASK SYSTEM

TRANSPORT CoC AMENDMENT REQUEST #3, REVISION 1

TRANSMITTAL LETTER + SEVEN ATTACHMENTS

1. WRITTEN RESPONSES TO NON-PROP RAIs
2. UPDATED NON-PROPRIETARY PROPOSED REVISED SAR PAGES
3. UPDATED MARKED-UP CoC
4. UPDATED REVISED CoC
5. AFFIDAVIT PURSUANT TO 10 CFR 2.390
6. WRITTEN RESPONSES TO PROPRIETARY RAIs
7. UPDATED PROPRIETARY PROPOSED REVISED SAR PAGES

27 JULY 2005

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Please contact us if you have any questions on or require further clarification of the information contained in the above attachments.

Sincerely,

Evan Rosenbaum

Evan Rosenbaum for Stefan Anton, Dr.-Ing.
Licensing Manager

Attachments: As Stated

cc: Mr. Meraj Rahimi, NRC

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Non-Proprietary Attachment 1

Written Responses to Non-Proprietary RAIs

(9 pages plus this cover sheet)

HI-STAR 100 TRANSPORT SYSTEM
DOCKET NO. 71-9261
TAC NO. L23651

REQUEST FOR ADDITIONAL INFORMATION

By application dated September 16, 2003, as supplemented by letter dated March 23, 2004, Holtec International (Holtec) requested approval of an amendment to Certificate of Compliance No. 71-9261, Revision 2, for the HI-STAR 100 Transportation Cask System. The enclosed request for additional information (RAI) identifies additional information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application for the amendment. The requested information is listed by chapter number, title, and section number in the applicant's safety analysis report. NUREG 1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," was used by the staff in its review of the application.

Each individual RAI describes information needed by the staff to complete its review of the application and/or the SAR and to determine whether that applicant has demonstrated compliance with the regulatory requirements.

Chapter 1 - Introduction

1-1 Revise the SAR definition of damaged fuel to match the currently approved CoC which incorporates the latest staff guidance contained in ISG-1, Rev. 1. Also, revise the definition of intact fuel accordingly.

This editorial change updates the SAR to match currently approved CoC, dated Sept. 24, 2003.

Holtec Response

A request to modify the definition of damaged fuel in the CoC such that the current SAR definition would no longer be in conflict has been submitted by Holtec letter 5014551 (dated December 30th, 2004), which is Holtec's fourth request to amend the HI-STAR System Part 71 CoC. NRC review of this fourth request is currently ahead of the review of the third request that this RAI is in reference to. Therefore, no changes are made in response to this RAI.

1-2 Specify whether or not ISG-11, Rev. 3, is desired rather than the presently referenced Rev. 2.

The staff has noted an inconsistency between the ongoing HI-STORM amendment and this Hi-STAR amendment with respect to the versions of ISG-11 that are referenced. The applicant must determine whether or not this apparent inconsistency has any material impact on the operation of the HI-STAR versus the HI-STORM.

Holtec Response

It is intended to use ISG-11 Revision 3 throughout. This has been corrected in a revision to the proposed SAR. Proposed SAR sections 1.2.1.6, 1.2.3.5, 3.0, 3.3, 3.6 and 3.7 have been affected by this correction.

1-3 Label the table on Page 1.2-51 in the SAR.

This table is referred to on Page 1.2-2 in Chapter 1 of the SAR as Table 1.2.18. The same table number with a description needs to be provided on Page 1.2-51.

Holtec Response

The correct label for this table has already been added by Holtec letter 5014551 (dated December 30th, 2004), which is Holtec's fourth request to amend the HI-STAR System Part 71 CoC. NRC review of this fourth request is currently ahead of the review of the third request that this RAI is in reference to. Therefore, no changes are made in response to this RAI.

1-4 Clarify the difference in the proposed CoC minimum pitch allowed of 9.158 inches and Drawing 3927, sheet 3, Rev. 6 allowed pitch of 9.218 +/- 0.06 inches.

This information is needed to assure compliance with 10 CFR 71.47 and 71.51.

Holtec Response

There is no difference between the minimum pitch in the proposed CoC and the pitch specification on drawing 3927 (i.e., 9.218" - 0.06" = 9.158"). Note that only the minimum pitch is specified as a requirement in the CoC, as a result of the criticality calculations.

Chapter 2 - Structural

No additional information is needed.

Chapter 3 - Thermal

3-1 Explain why in the MPC-32 ANSYS thermal model (and other MPC types), the internal basket panels surrounding any given fuel cell are represented by a single material (e.g., defined material no. 2 in the MPC-32 ANSYS model).

According to the SAR, the internal basket panels are modeled as orthotropic material with along-panel and through-panel defined thermal conductivities. As specified in the applicant's ANSYS thermal model, a material oriented in the X-direction would be using the correct associated thermal conductivities (along-panel for the X-direction and through-panel in the Y-direction). However, the same material oriented in the Y-direction would be incorrectly using these thermal conductivities (i.e., along-panel thermal conductivity would be used instead of through-panel thermal conductivity, etc.) Based on the applicant's ANSYS thermal model defined coordinate system, different materials should be used for the internal panels to correctly capture the orthotropic nature of this material.

This information is needed to assure compliance with 10 CFR 71.7 and 71.33.

Holtec Response

In ANSYS, material properties are applied to the elements using the element coordinate system. The ANSYS thermal model employs a local (x,y) co-ordinate system for defining the element coordinate system of the composite (i.e., equipped with neutron absorber and sheathing) basket panels oriented in the Y-direction. For panels oriented in the X-direction, the global (x,y) coordinate system is used to define the element co-ordinate system. For defining the properties of panels oriented in the Y-direction, the element coordinate system is defined by a 90° counter-clockwise rotated local coordinate system relative to the global coordinate system. In this manner a single material definition for the composite basket panels, with unequal through thickness and along the panel conductivities, is used appropriately for defining the non-isotropic panel conductivities. We note that this finite-element modeling approach has been employed in all HI-STAR 100 System analyses since the original CoC was issued in 1999.

3-2 Clarify whether the Rayleigh effect is credited in the thermal analysis of the HI-STAR 100 system.

Page 3.4-13 of the SAR states that for conservatism, the heat dissipation enhancement due to Rayleigh effect is ignored. However, page 3.4-30 of the SAR states that the Rayleigh effect thermal conductivity multipliers are unchanged in this analysis, giving the impression that in fact, helium gas conductivity was modified using some kind of multipliers. If Rayleigh effect is considered in the thermal analysis, justification and validation should be provided.

This information is needed to assure compliance with 10 CFR 71.7 and 71.33.

Holtec Response

For conservatively maximizing HI-STAR normal transport temperatures, convection heat dissipation in the basket peripheral spaces (Rayleigh effect) is ignored. In the evaluation of helium dilution by high molecular weight gases (fission gas releases from hypothetical rupture of all fuel rods), however, the increase in heat transfer due to a substantial rise in gas density is included. As this is a study to evaluate the effects of such a hypothetical rupture, it has no impact on actual operating temperatures. The SAR text in Section 3.4 has been revised to clarify the basket periphery heat transfer assumptions.

3-3 Correct the apparent referencing errors in the SAR as described below.

a. Page 3.4-31 states that low heat emitting fuel characteristics (including burnup and cooling time limits imposed on this class of fuel) are presented in Table 2.1.6. Table 2.1.6 does not contain this information.

a. In Section 3.4.4.1, references are made to Tables 4.4.6 and 4.4.7 which do not exist.

b. In Page 3.4-34, a reference is made to Holtec Drawing 1809 which does not exist.

This information is needed to assure compliance with 10 CFR 71.7 and 71.33.

Holtec Response

a. This is an editorial error. The correct reference is Table 1.2.23.
b. These are editorial errors. The correct references are Tables 3.4.5 and 3.4.6.
c. This is an editorial error. The correct reference is Drawing 3930.

The SAR text in Section 3.4 has been corrected to remove these errors.

3-4 Provide the maximum and allowable temperatures of other devices and/or equipment (namely, personnel barrier, tie-down system, support cradle, etc.) installed on the HI-STAR 100 system under normal conditions of transport.

The use of this equipment may have an adverse impact in the calculated temperatures by adding additional resistance to the dissipation of heat from the transport overpack to the environment.

This information is needed to assure compliance with 10 CFR 71.7, 71.33, and 71.43(g).

Holtec Response

The maximum and allowable temperatures of devices used to secure the HI-STAR to a transport vehicle (truck or rail car) are provided below:

Item	Maximum Temperature (°F)	Allowable Temperature (°F)
Tie Down Slings	222	250
Support Cradle	222	250
Personnel Barrier	168	185 (10 CFR 71 limit)

The tie down slings are long narrow bands that wrap 180° around the HI-STAR overpack belly. Because of a very low surface coverage the HI-STAR thermal performance is insensitive to presence of tie down devices. The support cradle is fabricated using high heat dissipating materials (structural steels) in direct contact with the underside of the HI-STAR overpack. As such the support cradle aids in the dissipation of heat.

The personnel barrier is a cage-type structure engineered with large openings to allow unrestricted access to ambient air. The pertinent personnel barrier specifications that ensure adequate cooling of the HI-STAR overpack are tabulated below:

Minimum Flow Opening Size	1 inch
Minimum Percentage Open Area	85%
Maximum Percentage Area Blocked by Solid Support Structure	3%

The tie down and cradle temperature limits and personnel barrier specifications define appropriate requirements for transport vehicle design. The personnel barrier specifications minimize airflow resistance. The employ of relatively large 1 inch openings for airflow ensures that the characteristic length scale for airflow resistance (the boundary layer thickness[a]) is bounded by a liberal margin.

Chapter 4 - Containment

4-1 Provide the references for both the normal transport conditions and the hypothetical accident conditions for the following parameters listed in Table 4.2.12 of the SAR:

- Upstream pressure
- Downstream pressure
- Temperature

This information is needed to assure compliance with 10 CFR 71.51.

Holtec Response

The upstream pressure for normal conditions of transport is taken as the maximum normal operating pressure in the most restrictive MPC from Table 3.4.15. The maximum pressure occurs in the MPC-32 with 3% rod rupture and is 89.3 psig (104 psia). The upstream pressure for accident conditions is assumed to be the accident condition design pressure of 200 psig (214.7 psia). The downstream pressure for normal and accident conditions is the pressure outside the HI-STAR transport overpack, which is assumed to be atmospheric pressure (14.7 psia). The maximum temperature for normal conditions of transport is assumed to be 530K = 494.6°C. This value bounds the MPC Bulk Cavity Temperature for normal operating conditions for all MPCs reported in Table 3.4.15. The maximum temperature specified for accident conditions is the maximum allowable accident condition peak cladding temperature of 1058°C = 843K.

4-2 Clarify how the normal transport condition temperature listed in Table 4.2.12 of the SAR was determined.

[a] Boundary layer thickness for natural convection cooling of heated surfaces is approximately 0.4 inch (McAdams, "Heat Transmission").

This temperature appears to be inconsistent with assumption 15 on page 4.2-3 of the SAR, which states:

> "The average cavity temperature for all analyses is conservatively assumed to be the design basis peak cladding temperature."

In Chapter 3 of the SAR, the design basis peak cladding temperature for normal transport conditions is listed as 752 degrees Fahrenheit. (Reference Table 3.4.10). This value was not used for the containment analysis. Explain this discrepancy.

This information is needed to assure compliance with 10 CFR 71.51.

<u>Holtec Response</u>

Please see the response to question 4-1 for determination of the normal transport condition temperature presented in Table 4.2.12. The assumption in Section 4.2 has been modified.

4-3 Clarify how the normal transport condition upstream pressure listed in Table 4.2.12 of the SAR was determined.

This pressure appears to be inconsistent with assumption 14 on page 4.2-3 of SAR, which states:

> "... the internal pressure of the overpack is conservatively assumed to be larger than the maximum internal pressure of all MPC types determined in Chapter 3."

In Chapter 3 of the SAR, the maximum internal pressure for MPC-32 is listed as 89.3 psig (104 psia). Table 4.2.12 states the upstream pressure for normal conditions is 104 psia, which is not larger than the value stated in Chapter 3. Explain this discrepancy.

This information is needed to assure compliance with 10 CFR 71.51.

<u>Holtec Response</u>

This assumption has been modified to say "... larger than or equal to the maximum internal pressure ..."

4-4 Revise the statement "Isotopes which contribute greater than 0.01% but have a radiological half-life less than 10 days are neglected" on Page 4.2-5 of the SAR.

The analysis in Holtec Report No: HI-971780, "*Containment Analysis for the HI-STAR 100,*" shows that the parent isotopes of the short-lived radionuclides (e.g., Ba-137m and Rh-106) are accounted for in the A2 calculations. Isotopes that have half-lives less than (1) 10 days, and (2) the half-life of their parent isotope may be considered to be in secular equilibrium with their parents. According to 10 CFR 71, Appendix A, III, isotopes in secular equilibrium with their parent isotopes may be treated as a single radionuclide, and the A2 value to be taken into account should correspond to the parent nuclide (e.g.

Cs-137 and Ru-106) of the decay chain. Since the parent isotopes are accounted for in the A2 determinations, the short-lived isotopes are not neglected, as Page 4.2-5 indicates.

This information is needed to assure compliance with 10 CFR 71, Appendix A, III.

Holtec Response

Agreed. This paragraph has been modified to make clear that those isotopes that have no A_2 value in Table A-1 from Appendix A of 10CFR71, have a half life less than ten days and have a half-life shorter than their parent nuclide (i.e., are in secular equilibrium with their parent nuclide), are in accordance with 10CFR71, Appendix A, III, treated as a single radionuclide along with their parent nuclide.

4-5 Revise the column headings in Table 4.2.2 (Pages 4.2-16 through 4.2-20 of the SAR) for consistency.

On Page 4.2-16, the second column is titled "PWR MPCs;" on Pages 4.2-17 through 4.2-20, it is titled "MPC-24."

Holtec Response

Agreed, these column headings have been changed.

Chapter 5- Shielding

5-1 Provide detailed justification for not explicitly analyzing the MPC-32 for azimuthal peaking as stated on page 5.4-3.

This information is needed to assure compliance with 10 CFR 71.47 and 71.51.

Holtec Response

Section 5.4.1 provides a detailed analysis of the radiation streaming through the ribs and the pocket trunnions in the HI-STAR 100. The effect of azimuthal positioning of fuel in the basket is also inherently accounted for in this analysis. Results are presented for the MPC-24 and MPC-68 in Tables 5.4.14 and 5.4.15 as discussed in Section 5.4.1. All other dose rates reported in Sections 5.1 and 5.4 are surface average dose rates. Section 5.5, Regulatory Compliance, presents dose rates that are not surface average but rather local peak dose rates taking into account radiation streaming through the pocket trunnions and radial steel ribs by using the peak-to-average values calculated in Section 5.4.1.

The last paragraph in Section 5.4.1 states that the MPC-32 was not explicitly analyzed for azimuthal peaking. The meaning of this statement is that peak-to-average values were not explicitly calculated for the MPC-32 and therefore are not reported in Section 5.4.1. However, in the determination of the dose rates in Section 5.5, Regulatory Compliance, the peak-to-average values for the MPC-24 were used for the MPC-32 to determine the peak dose rates for the MPC-32. This point is not stated in Section 5.4.1. This approach is acceptable because both the

MPC-24 and MPC-32 contain PWR fuel and the MPC-32 has a pattern which is more uniform and tightly packed and as a result the effect of azimuthal variation on the peak-to-average values in the MPC-24 should be larger and more severe than in the MPC-32. In conclusion, peak-to-average values calculated for the MPC-24 were used for the MPC-32 since peak-to-average values were not explicitly calculated for the MPC-32.

The last paragraph in Section 5.4.1 has been modified to read as follows.

"The MPC-32 was not explicitly analyzed to determine peak-to-average ratios. This is acceptable because the peaking outside the HI-STAR for the MPC-32 will be similar if not smaller than in the MPC-24 due to the fact that the fuel assemblies in the MPC-24 are not as closely positioned to each other as in the MPC-32. Section 5.5, Regulatory Compliance, presents results which take into account peaking due to radiation streaming or azimuthal variation. For the MPC-32, the peak-to-average values calculated for the MPC-24 were used."

5-2 Figures 5.3.4 and 5.3.5 show the MPC-32 basket cell as modeled in MCNP, one with Boral on all sides and one with Boral on no sides. Clarify why there is no model with the Boral on two sides or one side as shown in Drawing 3927, Sheet 3, Rev. 6.

This information is needed to assure compliance with 10 CFR 71.47 and 71.51.

Holtec Response

Figure 5.3.4 is the figure showing the basket for the MPC-32 with as modeled dimensions. Figure 5.3.5 is a figure for the MPC-24. Figure 5.3.4 shows an interior basket cell that has Boral and sheathing on all four sides. The peripheral cells were modeled correctly without Boral on the exterior cell wall closest to the MPC shell. In these peripheral locations the dimensions of the model shown in Figure 5.3.4 are correct. The only difference is that one or more Boral panels and associated sheathing, as depicted in the figure, were eliminated from the model as appropriate. The resulting MCNP model of the MPC-32 basket has no Boral panels or sheathing on the exterior basket walls. Figure 1.2.4 in Chapter 1 shows a drawing of the MPC-32 with Boral panels in the correct location. The full MCNP model of the MPC-32 replicated the Boral panel positions as shown in this figure.

5-3 Figure 5.3.9 displays a detailed cross sectional view of the HI-STAR 100 overpack with the MPC-24 (showing the thickness of the MPC shell and overpack as modeled in MCNP). Provide a similarly detailed view of the HI-STAR 100 with the MPC-32 on Figure 5.3.1.

This information is needed to assure compliance with 10 CFR 71.47 and 71.51.

Holtec Response

Figure 5.3.9 provides the detail of the HI-STAR 100 overpack as it was modeled in MCNP. The dimensions shown in the figure were used for all MPCs (MPC-24, MPC-32, and MPC-68). The MPC-24 is only shown as a representative basket. The figure caption has been modified to state that the MPC-24 is shown only for illustrative purposes only.

Chapter 7 - Operating Procedures

Pending the resolution of burnup verification measurements, additional information may be needed.

Chapter 8 - Acceptance Criteria Maintenance Procedures

8-1 Provide an addition to the Acceptance Criteria to include the Holtec QA/QC requirements for the testing of neutron absorber material(s). The appropriate procedures may be incorporated by reference as was recently proposed for the Hi-Storm Amendment 2, Rev. 2 (presently under review by the NRC staff). In that amendment, Hi-Storm SAR Section 9.1.5.3 was incorporated into the Technical specifications by reference.

The basis for this change is the recognition that neutron absorber materials are proprietary materials. As such, these materials are not subject to the uniform production and quality control standards that exist for ASME Code materials. Additionally, that there is no reasonable manner in which to verify the performance of these materials during service. The function they perform is of high importance; eliminating the possibility of an inadvertent criticality. Consequently, the NRC staff finds that the production and quality control methods and requirements of these materials need to be better formalized. In this manner, therefore, no changes to the materials production methods may occur unless such (proposed) changes are first subjected to an independent review.

Holtec Response

The testing requirements in the second and third paragraphs of SAR Section 8.1.5.3 have been incorporated by reference into the CoC, as requested. A note has been added to the SAR to identify these paragraphs as being incorporated by reference into the CoC.

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Non-Proprietary Attachment 2

Updated Non-Proprietary Proposed Revised SAR Pages

(342 pages plus this cover sheet)

1.2 PACKAGE DESCRIPTION

1.2.1 Packaging

The HI-STAR 100 System consists of an MPC designed for BWR or PWR spent nuclear fuel, an overpack that provides the containment boundary and a set of impact limiters that provide energy absorption capability for the normal and hypothetical accident conditions of transport. Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. This discussion is supplemented by a set of drawings in Section 1.4. Section 1.3 provides the HI-STAR 100 design code applicability and details any alternatives to the ASME Code.

Before proceeding to present detailed physical data on HI-STAR 100, it is contextual to summarize the design attributes that set it apart from the prior generation of spent fuel transportation packages.

There are several features in the HI-STAR 100 System design that increase its effectiveness with respect to the safe transport of spent nuclear fuel (SNF). Some of the principal features of the HI-STAR 100 System that enhance its effectiveness are:

- the honeycomb design of the MPC fuel basket
- the effective distribution of neutron and gamma shielding materials within the system
- the high heat rejection capability
- the structural robustness of the multi-shell overpack construction

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flanged plate weldment where all structural elements (box walls) are arrayed in two orthogonal sets of plates. Consequently, the walls of the cells are either completely coplanar (no offset) or orthogonal with each other. There is complete edge-to-edge continuity between contiguous cells.

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass over the body of the basket (in contrast to the "box and spacer disk" construction where the support plates are localized mass points). Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform (box and spacer disk) basket. In other words, the honeycomb basket is a more effective radiation attenuation device.

The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the HI-STAR 100 MPC an effective heat rejection device.

The multi-layer shell construction in the overpack provides a natural barrier against crack propagation in the radial direction across the overpack structure. If, during a hypothetical accident (impact) event, a crack was initiated in one layer, the crack could not propagate to the adjacent layer. Additionally, it is highly unlikely that a crack would initiate as the thinner layers are more ductile than a thicker plate.

In this Safety Analysis Report the HI-STAR 100 System design is demonstrated to have predicted responses to accident conditions that are clearly acceptable with respect to certification requirements for post-accident containment system integrity, maintenance of subcriticality margin, dose rates, and adequate heat rejection capability. Table 1.2.18 presents a summary of the HI-STAR 100 System performance against these aspects of post-accident performance at two levels. At the first level, the integrity of the MPC boundary prevents release of radioactive material or helium from the MPC, and ingress of moderator. The integrity of the MPC is demonstrated by the analysis of the response of this high quality, ASME Code, Section III, Subsection NB-designed, pressure vessel to the accident loads while in the overpack. With this demonstration of MPC integrity, the excellent performance results listed in the second column of Table 1.2.18 constitutes an acceptable basis for certification of the HI-STAR 100 System for the safe transport of spent nuclear fuel. However, no credit is taken for MPC integrity for certification of the HI-STAR 100 System for the transport of intact or damaged fuel assemblies. Credit is only taken for the additional containment boundary of the MPC-68F and MPC-24EF for the transport of fuel classified as fuel debris in order to meet the requirements of 10 CFR 71.63(b).

The HI-STAR 100 System provides a large margin of safety. The third column in Table 1.2.18 summarizes the performance if the MPC is postulated to suffer gross failure in the post-accident analysis. Even with this postulated failure, the performance of the HI-STAR 100 System is acceptable for the transport of intact and damaged fuel assemblies, showing the defense-in-depth methodology incorporated into the HI-STAR 100 System.

The containment boundary of the HI-STAR 100 System is shown to satisfy the special requirements of 10CFR71.61 for irradiated nuclear fuel shipments.

To meet the requirements of 10CFR71.63(b) for plutonium shipments, which is considered applicable for the transport of fuel classified as fuel debris, double containment is provided by the containment boundary of the overpack and the secondary containment boundary of the MPC-68F and MPC-24EF, serving as a separate inner container.

1.2.1.1 Gross Weight

The gross weight of the HI-STAR 100 System depends on which of the MPCs is loaded into the HI-STAR 100 overpack for shipment. Table 2.2.1 summarizes the maximum calculated component weights for the HI-STAR 100 overpack, impact limiters, and each MPC loaded to maximum capacity with design basis SNF. The maximum gross transport weight of the HI-STAR 100 System is to be marked on the packaging nameplate.

1.2.1.2 Materials of Construction, Dimensions, and Fabrication

All materials used to construct the HI-STAR 100 System are ASME Code materials, except the neutron shield, neutron poison, optional aluminum heat conduction elements, thermal expansion foam, seals, pressure relief devices, aluminum honeycomb, pipe couplings, and other material classified as Not Important to Safety. The specified materials of construction along with outline dimensions for important-to-safety items are provided in the drawings in Section 1.4.

The materials of construction and method of fabrication are further detailed in the subsections that follow. Section 1.3 provides the codes applicable to the HI-STAR 100 packaging for materials, design, fabrication, and inspection, including NRC-approved alternatives to the ASME Code.

1.2.1.2.1 HI-STAR 100 Overpack

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel. A single overpack design is provided that is capable of transporting each type of MPC. The inner diameter of the overpack is approximately 68-3/4 inches and the height of the internal cavity is approximately 191-1/8 inches. The overpack inner cavity is sized to accommodate the MPCs. The outer diameter of the overpack is approximately 96 inches and the height is approximately 203-1/4 inches.

Figure 1.2.1 provides a cross sectional elevation view of the overpack containment boundary. The overpack containment boundary is formed by a steel inner shell welded at the bottom to a bottom plate and, at the top, to a heavy top flange with a bolted closure plate. Two concentric grooves are machined into the closure plate for the seals. The closure plate is recessed into the top flange and the bolted joint is configured to protect the closure bolts and seals in the event of a drop accident. The closure plate has test and vent ports that are closed by a threaded port plug with a seal. The bottom plate has a drain port that is also closed by a threaded port plug with a seal. The containment boundary forms an internal cylindrical cavity for housing the MPC.

The outer surface of the overpack inner shell is buttressed with intermediate shells of gamma shielding that are installed in a manner to ensure a permanent state of contact between adjacent layers. Besides serving as an effective gamma shield, these layers provide additional strength to the overpack to resist puncture or penetration. Radial channels are vertically welded to the outside surface of the outermost intermediate shell at equal intervals around the circumference. These radial channels act as fins for improved heat conduction to the overpack outer enclosure shell surface and as cavities for retaining and protecting the neutron shielding. The enclosure shell is formed by welding enclosure shell panels between each of the channels to form additional cavities. Neutron shielding material is placed into each of the radial cavity segments formed by the radial channels, the outermost intermediate shell, and the enclosure shell panels. The exterior flats of the radial channels and enclosure shell panels form the overpack outer enclosure shell (Figure 1.2.2). Atop the outer enclosure shell, pressure relief devices (e.g., rupture disks) are positioned in a recessed area. The relief devices relieve internal pressure that may develop as a result of the fire accident and subsequent off-gassing of the neutron shield material. Within each radial channel, a layer of silicone sponge is positioned to act as a thermal expansion foam to compress as the neutron shield expands in the axial direction. Appendix 1.C provides material information on the thermal expansion foam. Figure 1.2.2 provides a mid-plane cross section view of the overpack, depicting the inner shell, intermediate shells, radial channels, outer enclosure shell, and neutron shield.

The exposed steel surfaces (except seal seating surfaces) of the overpack and the intermediate shell layers are coated to prevent corrosion. Coating materials are chosen based on the expected service conditions, considering the dual purpose certification status of the HI-STAR 100 System under 10 CFR 72 for spent fuel storage as well as transportation. The coatings applied to the overpack exposed exterior and interior surfaces are specified on the drawings in Section 1.4. The material data on the coatings is provided in Appendix 1.C. The inner cavity of the overpack is coated with a material appropriate to its high temperatures and the exterior of the overpack is coated with a material appropriate for fuel pool operations and environmental exposure. The coating applied to the intermediate shells acts as a surface preservative and is not exposed to the fuel pool or ambient environment.

Lifting trunnions are attached to the overpack top flange for lifting and rotating the cask body between vertical and horizontal positions. The lifting trunnions are located 180° apart in the sides of the top flange. On overpack serial numbers 1020-001 through 1020-007, pocket trunnions are welded to the lower side of the overpack 180° apart to provide a pivoting axis for rotation. The pocket trunnions are slightly off-center to ensure proper rotation direction of the overpack. As shown in Figure 1.1.4, the trunnions do no protrude beyond the cylindrical envelope of the overpack outer enclosure shell. This feature reduces the potential for direct impact on a trunnion in the event of an overpack side impact. After fabrication of HI-STAR overpack serial number 1020-007, the pocket trunnions were deleted from the overpack design.

1.2.1.2.2 Multi-Purpose Canisters

1.2.1.2.2.1 General Description

In this subsection, discussion of those attributes applicable to all of the MPC models is provided. Differences among the models are discussed in subsequent subsections. Specifications for the authorized contents of each MPC model, including non-fuel hardware and neutron sources are provided in Section 1.2.3.

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, a canister shell, a lid with vent and drain ports and cover plates, and a closure ring. The outer diameter of all MPCs and cylindrical height of each generic design MPC is fixed (see discussion in Subsection 1.2.1.2.2.3 regarding Trojan plant-specific MPCs). The number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. As the generic MPCs are interchangeable, they correspondingly have identical exterior dimensions. The outer dimension of the MPC is nominally 68-3/8 inches and the length is nominally 190-1/4 inches. Figures 1.2.3-1.2.5 depict the cross sectional views of the different MPCs. Drawings of the MPCs are provided in Section 1.4. Key system data for the HI-STAR 100 System are outlined in Tables 1.2.2 and 1.2.3.

The generic MPC-24/24E/24EF and Trojan plant MPC-24E/EF differ in construction from the MPC-32 and MPC-68/68F in one important aspect: the fuel cells are physically separated from one another by a flux trap between each cell for criticality control (Figures 1.2.3 and 1.2.4). All MPC baskets are formed from an array of plates welded to each other, such that a honeycomb structure is created that resembles a multi-flanged, closed-section beam in its structural characteristics.

The MPC fuel basket is positioned and supported within the MPC shell by a series of basket supports welded to the inside of the MPC shell. In the peripheral area created by the basket, the MPC shell, and the basket supports, optional aluminum heat conduction elements are installed in some early production MPC-68 and MPC-68F models (see Figure 1.2.3). These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design that allow a snug fit in the confined spaces and ease of installation. The heat conduction elements are along the full length of the MPC basket, except at the drain pipe location, to create a nonstructural thermal connection that facilitates heat transfer from the basket to the shell. In their operating condition, the heat conduction elements conform to, and contact the MPC shell and basket walls. In SAR Revision 10, a refined thermal analysis, described in Chapter 3, has allowed the elimination of these heat conduction elements from the MPC design, thus giving this design feature "optional" status.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit placement of the empty MPC into the overpack, and are considered non-structural, non-pressure retaining attachments to the MPC pressure boundary. The lifting lugs also serve to axially locate the MPC lid prior to welding. These internal lifting lugs are not used to handle a loaded MPC, since the MPC lid blocks access to the lifting lugs.

The top of the HI-STAR 100 MPC incorporates a redundant closure system. Figure 1.2.6 provides a sketch of the MPC closure details. The MPC lid is a circular plate (fabricated from one piece, or two pieces - split top and bottom) that is edge-welded to the MPC shell. If the two-piece lid design is employed, only the top piece is analyzed as part of the enclosure vessel pressure boundary. The bottom piece acts primarily as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld, as depicted on the MPC enclosure vessel drawing in Section 1.4. The MPC lid is equipped with vent and drain ports that are used to remove moisture and gas from the MPC and backfill the MPC with a specified pressure of inert gas (helium). The vent and drain ports are sealed closed by cover plates welded to the MPC lid before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and MPC lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the threaded holes in the MPC lid during transfer from the storage-only HI-STORM 100 System to the HI-STAR 100 overpack for transportation. Threaded insert plugs are installed to provide shielding when the threaded holes are not in use.

All MPCs are <u>designed</u> to handle intact fuel assemblies, damaged fuel assemblies, and fuel classified as fuel debris. Damaged fuel and fuel debris must be transported in damaged fuel containers or other approved damaged/failed fuel canister. At this time, only BWR damaged fuel and fuel debris from the Dresden Unit 1 and Humboldt Bay plants is certified for transportation in the MPC-68 and the MPC-68F. Similarly, only PWR damaged fuel and fuel debris from the

Trojan plant is certified for transportation in the Trojan plant-specific MPC-24E and the MPC-24EF. The definitions, and applicable specifications for all authorized contents, including the requirements for canning certain fuel, are provided in Subsection 1.2.3.

Intact SNF can be placed directly into the MPC. Damaged SNF and fuel debris must be placed into a Holtec damaged fuel container or other authorized canister for transportation inside the MPC and the HI-STAR 100 overpack. Figures 1.2.10 through 1.2.11 provide sketches of the containers authorized for transportation of damaged fuel and fuel debris in the HI-STAR 100 System. One Dresden Unit 1 Thoria rod canister, shown in Figure 1.2.11A, is also authorized for transportation in HI-STAR 100.

In order to qualify the MPC-68F and MPC-24EF shells as a secondary containment boundary for the transportation of Dresden Unit 1/Humboldt Bay and Trojan plant fuel debris, respectively, the MPC-68 and MPC-24E enclosure vessels have been slightly modified to further strengthen the lid-to-shell joint area. These fuel debris MPCs are given the "F" suffix (hence, MPC-68F and MPC-24EF)†. The differences between the standard and "F-model" MPC lid-to-shell joints are shown on Figure 1.2.17, and include a thickened upper shell, a larger lid-to-shell weld size, and a correspondingly smaller lid diameter. The design of the rest of the enclosure vessel is identical between the standard MPC and the "F-model" MPC.

The MPC-68F and MPC-24EF provide the separate inner container per 10CFR71.63(b) for the HI-STAR 100 System transporting fuel classified as fuel debris to ensure double containment. The overpack containment boundary provides the primary containment boundary.

1.2.1.2.2.2 MPC-24/24E/24EF

The MPC-24 is designed to transport up to 24 PWR intact fuel assemblies meeting the limits specified in Subsection 1.2.3. The MPC 24E is designed to transport up to 24 PWR intact and up to four PWR damaged fuel assemblies in damaged fuel containers. The MPC-24EF is designed to transport up to 24 PWR intact fuel assemblies and up to four PWR damaged fuel assemblies or fuel assemblies classified as fuel debris. At this time, however, generic PWR damaged fuel and fuel debris are not authorized for transportation in the MPC-24E/EF.

All MPC-24-series fuel baskets employ the flux trap design for criticality control, as shown in the drawings in Section 1.4. The fuel basket design for the MPC-24E is an enhanced MPC-24 basket layout designed to improve the fuel storage geometry for criticality control. The fuel basket design of the MPC-24EF is identical to the MPC-24E. The MPC-24E/EF basket designs also employ a higher ^{10}B loading than the MPC-24, as shown in Table 1.2.3. The differences between the MPC-24EF enclosure vessel design and the MPC-24/24E enclosure vessel are discussed in Subsection 1.2.1.2.2.1.

† The drawing in Section 1.4 also denotes an MPC-68FF fuel debris canister design. However, the MPC-68FF is not authorized for use in transportation under the HI-STAR 100 10 CFR 71 CoC.

1.2.1.2.2.3 Trojan Plant MPC-24E/EF

The Trojan plant MPC-24E and -24EF models are designs that have been customized for that plant's fuel and the concrete storage cask being used at the Trojan plant Independent Spent Fuel Storage Installation (ISFSI) (Docket 72-0017). The design features that are unique to the Trojan plant MPCs are specifically noted on the MPC enclosure vessel and MPC-24E/EF fuel basket drawings in Section 1.4. These differences include:

- a shorter MPC fuel basket and cavity length to match the shorter Trojan fuel assembly length
- shorter corner fuel storage cell lengths to accommodate the Trojan Failed Fuel Cans
- a different fuel storage cell and flux trap dimension in the corner cells to accommodate the Trojan Failed Fuel Cans
- a different configuration of the flow holes at the bottom of the fuel basket (rectangular vs. semi-circular)

All other design features in the Trojan MPCs are identical to the generic MPC-24E/EF design. The HI-STAR 100 overpack design has not been modified for the Trojan MPC design.

The technical analyses described in this SAR were verified in most cases to bound the Trojan-specific design features. Where necessary, Trojan plant-specific evaluations were performed and are summarized in the appropriate SAR section. To accommodate the shorter Trojan plant MPC length in a standard-length HI-STAR 100 overpack, a spacer was designed for installation into the overpack above the Trojan MPC (see Figure 1.1.5 and the drawing in Section 1.4) for transportation in the standard-length HI-STAR 100 overpack. This spacer prevents the MPC from moving more than the MPC was analyzed to move in the axial direction and serves to transfer the axial loads from the MPC lid to the overpack top closure plate within the limits of the supporting analyses. See Section 2.7.1.1 for additional discussion of the spacer used with the Trojan MPC design. Hereafter in this SAR, the Trojan plant-specific MPC design is only distinguished from the generic MPC-24E/EF design when necessary to describe unique evaluations performed for those MPCs.

1.2.1.2.2.4 MPC-32

~~NOTE: The MPC-32 is not certified for transportation at this time.~~

The MPC-32 is designed to transport up 32 PWR intact fuel assemblies meeting the specifications in Subsection 1.2.3. Damaged fuel and fuel debris are not permitted to be transported in the MPC-32. The MPC-32 enclosure vessel design is identical to the MPC-24/24E enclosure vessel design as shown on the drawings in Section 1.4. The MPC-32 fuel basket does not employ flux traps for criticality control. Credit for burnup of the fuel is taken in the criticality analyses for accident conditions and to meet the requirements of 10 CFR 71.55(b). Because the MPC is designed to preclude the intrusion of moderator under all normal and credible accident conditions of transport, the moderator intrusion condition analyzed as required by 10 CFR 71.55(b) is a non-mechanistic event for the HI-STAR 100 System.

1.2.1.2.2.5 MPC-68/68F

The MPC-68 is designed to transport up to 68 BWR intact fuel assemblies and damaged fuel assemblies meeting the specifications in Subsection 1.2.3. Zircaloy channels are permitted. At this time, only damaged fuel from the Dresden Unit 1 and Humboldt Bay plants is authorized for transportation in the MPC-68. The MPC-68F is designed to transport only fuel and other authorized material from the Dresden Unit 1 and Humboldt Bay plants meeting the specifications in Subsection 1.2.3. The sole difference between the MPC-68 and MPC-68F fuel basket design is a reduction in the required ^{10}B areal density in the Boral. A reduction in the required ^{10}B areal density of the Boral is possible for the MPC-68F due to limited types of fuel and low enrichments permitted to be transported in this MPC model. The differences between the MPC-68F enclosure vessel design and the MPC-68 enclosure vessel are discussed in Subsection 1.2.1.2.2.1.

1.2.1.2.2.6 Alloy X

The HI-STAR MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and aluminum vent and drain cap seal washers in all MPCs, and the aluminum heat conduction elements in the first several production units of MPC-68 and MPC-68F). No carbon steel parts are used in the design of the HI-STAR 100 MPC. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the HI-STAR MPCs. All structural components in a HI-STAR MPC will be fabricated of Alloy X, a designation that warrants further explanation.

Alloy X is a fictitious material that should be acceptable as a Mined Geological Depository System (MGDS) waste package and that meets the thermophysical properties set forth in this document.

At this time, there is considerable uncertainty with respect to the material of construction for an MPC that would be acceptable as a waste package for the MGDS. Candidate materials being considered for acceptability by the DOE include:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The DOE material selection process is primarily driven by corrosion resistance in the potential environment of the MGDS. As the decision regarding a suitable material to meet disposal requirements is not imminent, this application requests approval for use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this SAR) may be one of the following materials. Any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed below, except that all steel pieces comprising the MPC shell (i.e., the 1/2" thick cylinder) must be fabricated from the same Alloy X stainless steel type:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable for the entire group for the analysis in question. For example, when calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, we have defined a material, which is referred to as Alloy X, whose thermophysical properties, from the MPC design perspective, are the least favorable of the candidate materials group. The evaluation of the Alloy X constituents to determine the least favorable properties is provided in Appendix 1.A.

The Alloy X approach is conservative because no matter which material is ultimately utilized, the Alloy X approach guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

1.2.1.3 Impact Limiters

The HI-STAR 100 overpack is fitted with aluminum honeycomb impact limiters, termed AL-STAR™, one at each end, once the overpack is positioned and secured in the transport frame. The impact limiters ensure the inertia loadings during the normal and hypothetical accident conditions of transport are maintained below design levels. The impact limiter design is discussed further in Chapter 2 and drawings are provided in Section 1.4.

1.2.1.4 Shielding

The HI-STAR 100 System is provided with shielding to minimize personnel exposure. The HI-STAR 100 System will be transported by exclusive use shipment to ensure the external radiation requirements of 10CFR71.47 are met. During transport, a personnel barrier is installed to restrict access to the overpack to protect personnel from the HI-STAR 100 exterior surface temperature in accordance with 10CFR71.43(g). The personnel barrier provides a stand-off equal to the exterior radial dimension of the impact limiters. Figure 1.2.8 provides a sketch of the personnel barrier being installed.

The initial attenuation of gamma and neutron radiation emitted by the radioactive spent fuel is provided by the MPC fuel basket structure built from inter-welded plates and Boral neutron poison panels with sheathing attached to the fuel cell walls. The MPC canister shell, baseplate, and lid provide additional thicknesses of steel to further reduce gamma radiation and, to a smaller extent, neutron radiation at the outer MPC surfaces. No shielding credit is taken for the aluminum heat conduction elements installed in some of the early production MPC-68 and MPC-68F units.

The primary HI-STAR 100 shielding is located in the overpack and consists of neutron shielding and additional layers of steel for gamma shielding. Neutron shielding is provided around the outside circumferential surface of the overpack. Gamma shielding is provided by the overpack inner, intermediate and enclosure shells with additional axial shielding provided by the bottom plate and the top closure plate. During transport, the impact limiters will provide incremental gamma shielding and provide additional distance from the radiation source at the ends of the package. An additional circular segment of neutron shielding is contained within each impact limiter to provide neutron attenuation.

1.2.1.4.1 Boral Neutron Absorber

Boral is a thermal neutron poison material composed of boron carbide and aluminum alloy 1100. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The aluminum alloy 1100 is a lightweight metal with high tensile strength that is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal, and chemical environment of a nuclear reactor, spent fuel pool, or dry cask.

The documented historical applications of Boral, in environments comparable to those in spent fuel pools and fuel storage casks, dates to the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor [1.2.2]). Technical data on the material was first printed in 1949, when the report "Boral: A New Thermal Neutron Shield" was published [1.2.3]. In 1956, the first edition of the "Reactor Shielding Design Manual" [1.2.4], contains a section on Boral and its properties.

In the research and test reactors built during the 1950s and 1960s, Boral was frequently the material of choice for control blades, thermal-column shutters, and other items requiring very good thermal-neutron absorption properties. It is in these reactors that Boral has seen its longest service in environments comparable to today's applications.

Boral found other uses in the 1960s, one of which was a neutron poison material in baskets used in the shipment of irradiated, enriched fuel rods from Canada's Chalk River laboratories to Savannah River. Use of Boral in shipping containers continues, with Boral serving as the poison in many cask designs.

Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

Boral has been exclusively used in fuel storage applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and several unique characteristics, such as:

- The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.
- Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- Boral is stable, strong, durable, and corrosion resistant.

Boral absorbs thermal neutrons without physical change or degradation of any sort from the anticipated exposure to gamma radiation and heat. The material does not suffer loss of neutron attenuation capability when exposed to high levels of radiation dose.

Holtec International's QA Program ensures that Boral is manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR71, Subpart H and 10CFR72, Subpart G. Holtec International has procured over 200,000 panels of Boral from AAR Advanced Structures for over 20 projects. Boral has always been purchased with a minimum ^{10}B loading requirement. Coupons extracted from production runs were tested using the "wet chemistry" procedure. The actual ^{10}B loading, out of thousands of coupons tested, has never been found to fall below the design specification. The size of this coupon data base is sufficient to provide confidence that all future procurements will continue to yield Boral with full compliance with the stipulated minimum loading. Furthermore, the surveillance, coupon testing, and material tracking processes that have so effectively controlled the quality of Boral are expected to continue to yield Boral of similar quality in the future. Nevertheless, to add another layer of insurance, only 75% ^{10}B credit of the fixed neutron absorber is assumed in the criticality analysis.

The oxide layer that is created from the reaction of the outer aluminum cladding and the edges of the Boral panels with air and water provides a barrier to further reaction of the aluminum cladding with air or the spent fuel pool water during loading and unloading operations. However, with extended submergence in an MPC filled with water or in the plant's spent fuel pool, the hydrostatic pressure can drive water into the Boral core (comprised of particulate B_4C and aluminum powder) where previously unexposed aluminum powder may react with the water to create hydrogen. The rate of hydrogen generation and the total hydrogen generated is dependent on several variables:

- Aluminum particle size: Aluminum particle size in the Boral core and associated porosity affects the amount of aluminum available for reaction with water. Larger aluminum particles yield less surface area for reaction, but higher porosity for aluminum-water interaction; smaller aluminum particles yield more surface area for reaction, but lower porosity for aluminum-water reaction.

- Presence of trace impurities: The presence of trace impurities in the Boral core due to the manufacturing process (i.e., sodium hydroxide, boron oxide, and iron-oxide) can affect the rate of hydrogen production, both increasing and suppressing the reaction. Sodium dissolved in the water increases the pH and tends to increase the rate of hydrogen production. This is counteracted by the boron oxide, which hydrolyzes to boric acid (H_3BO_3) and reduces the rate of hydrogen production. Trace impurities do not affect the total amount of hydrogen generated.

- Pool water chemistry: Chemicals in the plant spent fuel pool water (e.g., copper, boron) can affect the rate of hydrogen production, both increasing (copper) and suppressing (boron) the reaction.

- MPC loading operations: Operating needs or preferences by individual utilities as to when, and for how long the MPC is kept at varying water depths in the spent fuel pool, and how long the MPC is kept filled with water outside the spent fuel pool can affect the amount of aluminum in the Boral core that may be exposed to water.

Due to the variability in hydrogen generation from the Boral-water reaction, the operating procedures in Chapter 7 require monitoring for combustible gases and either exhausting or purging the space beneath the MPC lid during loading and unloading operations when an ignition event could occur (i.e., when the space beneath the MPC lid is open to the welding or cutting operation).

1.2.1.4.2 Holtite-A™ Neutron Shielding

The specification for the overpack and impact limiter neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation and associated neutron capture to appropriate levels;

- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;

- Stability of the homogeneous nature of the shielding material matrix;

- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and

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

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

Holtite-A is the only approved neutron shield material that fulfills the aforementioned criteria. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal B_4C loading of 1 weight percent for the HI-STAR 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

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

Density

The nominal specific gravity of Holtite-A is 1.68 g/cm^3 as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm^3. The density used for the shielding analysis is assumed to be 1.61 g/cm^3 to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The nominal weight concentration of hydrogen is 6.0%. However, all shielding analyses conservatively assume 5.9% hydrogen by weight in the calculations.

Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (nominal) weight concentration. Holtite-A may be specified with a B_4C content of up to 6.5 weight percent. For the HI-STAR 100 System, Holtite-A is specified with a nominal B_4C weight percent of 1%.

Design Temperature

The design temperature of Holtite-A is set at 300°F. The maximum spatial temperature of Holtite-A under all normal operating conditions must be demonstrated to be below this design temperature.

Thermal Conductivity

It is evident from Figure 1.2.2 that Holtite-A is directly in the path of heat transmission from the inside of the overpack to its outside surface. For conservatism, however, the design basis thermal conductivity of Holtite-A under heat rejection conditions is set equal to zero. The reverse condition occurs under a postulated fire event when the thermal conductivity of Holtite-A aids in the influx of heat to the stored fuel in the fuel basket. The thermal conductivity of Holtite-A is conservatively set at 1 Btu/hr-ft-°F for all fire accident analyses.

The Holtite-A neutron shielding material is stable at normal design temperatures over the long term and provides excellent shielding properties for neutrons.

1.2.1.4.3 Gamma Shielding Material

For gamma shielding, HI-STAR 100 utilizes carbon steel in plate stock form. Instead of utilizing a thick forging, the gamma shield design in the HI-STAR 100 overpack borrows from the concept of layered vessels from the field of ultra-high pressure vessel technology. The shielding is made from successive layers of plate stock. The fabrication of the shell begins by rolling the inner shell plate and making the longitudinal weld seam. Each layer of the intermediate shells is constructed from two halves. The two halves of the shell are precision sheared, beveled, and rolled to the required radii. The two halves of the second layer are wrapped around the first shell. Each shell half is positioned in its location and while applying pressure using a specially engineered fixture, the halves are tack welded. The beveled edges to be joined are positioned to make contact or have a slight gap. The second layer is made by joining the two halves using two longitudinal welds. Successive layers are assembled in a like manner. Thus, the welding of every successive shell provides a certain inter-layer contact (Figure 1.2.7).

A thick structural component radiation barrier is thus constructed with four key features, namely:

- The number of layers can be increased as necessary to realize the required design objectives.
- The layered construction is ideal to stop propagation of flaws.
- The thinner plate stock is much more ductile than heavy forgings used in other designs.
- Post-weld heat treatment is not required by the ASME Code, simplifying fabrication.

1.2.1.5 Lifting and Tie-Down Devices

The HI-STAR 100 overpack is equipped with two lifting trunnions located in the top flange. The lifting trunnions are designed in accordance with 10CFR71.45, NUREG-0612 [1.2.11], and ANSI N14.6 [1.3.3], manufactured from a high strength alloy, and are installed in threaded openings. The lifting trunnions may be secured in position by optional locking pads, shaped to make conformal contact with the curved overpack. Once the locking pad is bolted in position, the inner diameter is sized to restrain the trunnion from backing out. The two off-center pockets located near the overpack bottom plate on overpack serial numbers 1020-001 through 1020-007 are pocket trunnions. The pocket trunnions were eliminated from the design after serial number 1020-007 was fabricated and are no longer considered qualified tie-down devices. However, the pocket trunnions on these overpacks may still be used for normal handling activities such as upending and downending.

The lifting, upending, and downending of the HI-STAR 100 System requires the use of external handling devices. A lifting yoke is utilized when the cask is to be lifted or set in a vertical orientation. For those overpacks that have been fabricated with the pocket trunnions, transport and rotation cradles may include rotation trunnions that interface with the pocket trunnions to provide a pivot axis. A lift yoke may be connected to the lifting trunnions and the crane hook used for upending or downending the HI-STAR 100 System by rotating on the pocket trunnions for these overpacks. For those overpacks fabricated without pocket trunnions, the overpack must be transferred into the transport saddle with appropriate lift rigging. If an overpack having pocket trunnions is secured to the transport vehicle without engaging the pocket trunnions, plugs are required to be installed in the pocket to provide radiation shielding (see the overpack drawing in Section 1.4).

For transportation, the HI-STAR 100 System is engineered to be mounted on a transport frame secured to the transporter bed. Figure 1.2.8 provides a sketch of the HI-STAR 100 System secured for transport and the drawing in Section 1.4 provides additional details. The transport frame has a lower saddle with attachment points for belly slings around the cask body designed 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 loads into the cradle structure. See Section 2.5 for discussion of the qualification of tie-down devices.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered from the HI-STAR overpack. For users of the HI-STORM 100 Dry Storage System, MPC handling operations are performed using a HI-TRAC transfer cask of the HI-STORM 100 System (Docket No. 72-1014). The HI-TRAC transfer cask allows the sealed MPC loaded with spent fuel to be transferred from the HI-STORM 100 overpack (storage-only) to the HI-STAR 100 overpack, or vice versa. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6 and are plugged during transportation to prevent radiation streaming.

1.2.1.6 Heat Dissipation

The HI-STAR 100 System can safely transport SNF by maintaining the fuel cladding temperature below the limits specified in Table 1.2.3 for normal and accident conditions. These limits have been established consistent with the guidance in NRC Interim Staff Guidance (ISG) document No. 11, Revision ~~2~~*3* (Ref. [1.2.14]). The temperature of the fuel cladding is dependent on the decay heat and the heat dissipation capabilities of the cask. The total heat load per BWR and PWR MPC is identified in Table 1.2.3. The SNF decay heat is passively dissipated without any mechanical or forced cooling.

The HI-STAR 100 System must meet the requirements of 10CFR71.43(g) for the accessible surface temperature limit. To meet this requirement the HI-STAR 100 System is shipped as an exclusive use shipment and includes an engineered personnel barrier during transport.

The primary heat transfer mechanisms in the HI-STAR 100 System are conduction and surface radiation.

The free volume of the MPC and the annulus between the external surface of the MPC and the inside surface of the overpack containment boundary are filled with 99.995% pure helium gas during fuel loading operations. Table 1.2.3 specifies the acceptance criteria for helium fill pressure in the MPC internal cavity. Besides providing an inert dry atmosphere for the fuel cladding, the helium also provides conductive heat transfer across any gaps between the metal surfaces inside the MPC and in the annulus between the MPC and overpack containment boundary. Metal conduction transfers the heat throughout the MPC fuel basket, through the MPC aluminum heat conduction elements (if installed) and shell, through the overpack inner shell, intermediate shells, steel radial connectors and finally, to the outer neutron shield enclosure shell. The most adverse temperature profiles and thermal gradients for the HI-STAR 100 System with each of the MPCs are discussed in detail in Chapter 3. The thermal analysis in Chapter 3 no longer takes credit for the aluminum heat conduction elements and they have been designated as optional equipment.

1.2.1.7 Coolants

There are no coolants utilized in the HI-STAR 100 System. As discussed in Subsection 1.2.1.6 above, helium is sealed within the MPC internal cavity. The annulus between the MPC outer surface and overpack containment boundary is also purged and filled with helium gas.

1.2.1.8 Pressure Relief Systems

No pressure relief system is provided on the HI-STAR 100 packaging containment boundary.

The sole pressure relief devices are provided in the overpack outer enclosure (Figure 1.1.4). The overpack outer enclosure contains the neutron shield material. Normal loadings will not cause the rupture disks to open. The rupture disks are installed to relieve internal pressure in the neutron shield cavities caused by the fire accident. The overpack outer enclosure is not designed as a pressure vessel. Correspondingly, the rupture disks are designed to open at relatively low pressures as stated below.

Relief Device location	Set pressure, psig
Overpack outer enclosure	30, +/- 5

1.2.1.9 Security Seal

The HI-STAR 100 packaging provides a security seal that while intact, provides evidence that the package has not been opened by unauthorized persons. When installed, the impact limiters cover all penetrations into the HI-STAR 100 packaging containment boundary. Therefore, the security seal is placed to ensure that the impact limiters are not removed which thereby ensures that the package has not been opened. As shown on the HI-STAR transport assembly drawing in Section 1.4, security seals are provided on one impact limiter attachment bolt on the top impact limiter and through two adjacent bolts on the bottom impact limiter. A hole is provided in the head of the bolt and the impact limiter. Lockwire shall be threaded through the hole and joined with a security seal.

1.2.1.10 Design Life

The design life of the HI-STAR 100 System is 40 years. This is accomplished by using materials of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 8, is also implemented to ensure the HI-STAR 100 System will exceed its design life of 40 years. The design considerations that assure the HI-STAR 100 System performs as designed throughout the service life include the following:

HI-STAR Overpack

- Exposure to Environmental Effects
- Material Degradation
- Maintenance and Inspection Provisions

MPC

- Corrosion
- Structural Fatigue Effects
- Maintenance of Helium Atmosphere
- Allowable Fuel Cladding Temperatures
- Neutron Absorber Boron Depletion

1.2.2 Operational Features

Table 1.2.7 provides the sequence of basic operations necessary to load fuel and prepare the HI-STAR 100 System for transport. More detailed guidance for transportation-related loading, unloading, and handling operations is provided in Chapter 7 and is supported by the drawings in Section 1.4. A summary of the loading and unloading operations is provided below. Figures 1.2.9 and 1.2.16 provide a pictorial view of the loading and unloading operations, respectively.

1.2.2.1 Applicability of Operating Procedures for the Dual-Purpose HI-STAR 100 System

The HI-STAR 100 System is a dual-purpose system certified for use as a dry storage cask under 10 CFR 72 and a transportation package under 10 CFR 71. In addition, the MPC is certified for use under 10 CFR 72 in the storage-only HI-STORM 100 System (a ventilated concrete cask system). Therefore, it is possible that the HI-STAR 100 overpack and/or the MPC may be loaded, prepared, and sealed under the operating procedures for storage, delineated in the HI-STAR 100 storage FSAR (Docket 72-1008) or the HI-STORM 100 storage FSAR (Docket 72-1014). In those cases, the operating procedures governing MPC and overpack preparation for storage would apply. The MPC and HI-STAR 100 overpack, as applicable, must be confirmed to meet all requirements of the Part 71 Certificate of Compliance before being released for shipment.

For those instances where the MPC is being loaded and shipped off-site in a HI-STAR 100 overpack under 10 CFR 71 without first being deployed at an ISFSI (known as "load-and-go" operations), the operating procedures in Chapter 7 (and summarized below) apply for preparation of the MPC and HI-STAR overpack. For those cases where the MPC is transferred from storage in a HI-STORM overpack to a HI-STAR overpack for shipment, the operating procedures in Chapter 7 (and summarized below) govern the preparation activities for the HI-STAR overpack.

Loading Operations

At the start of loading operations, the overpack is configured with the closure plate removed. The lift yoke is used to position the overpack in the designated preparation area or setdown area for overpack inspection and MPC insertion. The annulus is filled with plant demineralized water and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with spent fuel pool water or plant demineralized water (borated as required for MPC-32). The overpack and MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielding lid (the MPC lid) is installed. The lift yoke is remotely engaged to the overpack lifting trunnions and is used to lift the overpack close to the spent fuel pool surface. The MPC lift bolts (securing the MPC lid to the lift yoke) are removed. As the overpack is removed from the spent fuel pool, the lift yoke and overpack are sprayed with demineralized water to help remove contamination.

The overpack is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the top flange of the overpack are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus (foreign material exclusion). If used, the Automated Welding System (AWS) is installed. The MPC water level is lowered slightly and the space under the MPC lid is purged or exhausted and monitoring is performed. The MPC lid is seal-welded using the AWS. Liquid penetrant examinations are performed on the root and final passes and ultrasonic examination is also performed on the MPC lid-to-shell weld or, in place of the ultrasonic examination, the weld may be inspected by multiple-pass liquid penetrant examination at approximately every 3/8 inch of weld depth. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid-to-shell) to verify weld integrity and to ensure that the leakage rates are within acceptance criteria. The MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line. At the appropriate time in the sequence of activities, based on the type of test performed (hydrostatic or pneumatic), a pressure test of the MPC enclosure vessel is performed.

The Forced Helium Dehydration (FHD) System is connected to the MPC and is used to remove residual water from the MPC and reduce the level of moisture in the MPC to acceptable levels. This is accomplished by recirculating dry, heated helium through the MPC cavity to absorb the moisture. When the helium exiting the MPC is determined to meet the required moisture limit, the MPC is considered sufficiently dried for transportation (see Section 3.4.1.1.16 for a description of the FHD System.

Following MPC drying operations, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer, provides an inert atmosphere for fuel cladding integrity, and provides the means of future leakage rate testing of the MPC enclosure vessel boundary welds. Cover plates are installed and seal-welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and/or final passes, depending on the number of weld passes required. That is, if only a single weld pass is required, only a final liquid penetrant examination is performed. The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.

The MPC closure ring is then placed on the MPC, aligned, tacked in place, and seal welded, providing redundant closure of the MPC enclosure vessel closure welds. Tack welds are visually examined, and the root and/or final welds (depending on the number of weld passes required) are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in the annulus is drained. The AWS is removed. The overpack closure plate is installed and the bolts are torqued. The overpack annulus is dried using the vacuum drying system (VDS).

If the MPC being transported is an "F-model" canister, a helium leakage test on the canister must be performed to confirm the integrity of the secondary containment boundary prior to backfilling the overpack annulus.

The overpack annulus is backfilled with helium gas for heat transfer and seal testing. Concentric metallic seals in the overpack closure plate prevent the leakage of the helium gas from the annulus and provide the containment boundary to the release of radioactive materials. The seals on the overpack vent and drain port plugs are leak tested along with the overpack closure plate inner seal. Cover plates with metallic seals are installed over the overpack vent and drain ports to provide redundant closure of the overpack penetrations. A port plug with a metallic seal is installed in the overpack closure plate test port to provide fully-redundant closure of all overpack penetrations.

The overpack is surveyed for removable contamination and secured on the transport vehicle with impact limiters installed, the security seals are attached, and the personnel barrier is installed. The HI-STAR 100 packaging is then ready for transport.

Unloading Operations

The HI-STAR 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC (if necessary), flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the overpack and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC overpressurization and thermal shock to the stored spent fuel assemblies.

After removing the impact limiters, the overpack and MPC are positioned in the designated preparation area. At the site's discretion, a gas sample is drawn from the overpack annulus and analyzed. The gas sample provides an indication of MPC enclosure vessel performance. The annulus is depressurized, the overpack closure plate is removed, and the annulus is filled with plant demineralized water. The annulus shield is installed to protect the annulus from debris produced from the lid removal process. Similarly, overpack top surfaces are covered with a protective fire-retarding blanket.

The Weld Removal System (WRS) is positioned on the MPC lid. The MPC closure ring is core drilled over the locations of the vent and drain port cover plates. The MPC closure ring and vent and drain port cover plates are core drilled to the extent necessary to allow access by the Remote Valve Operating Assemblies (RVOAs). Local ventilation is established around the vent and drain ports. The RVOAs are connected to allow access to the MPC cavity for re-flooding operations.

The MPC cavity gas is verified to be below an appropriate temperature (approximately 200°F) to allow water flooding. Depending on the time since initial fuel loading and the age and burnup of the contained fuel, mechanical cooling of the MPC cavity gas may or may not be required to ensure the cavity gas temperature meets the acceptance criterion. A thermal evaluation should be performed to determine the MPC bulk cavity gas temperature at the time of unloading. Based on that thermal evaluation, if the MPC cavity gas temperature does not already meet the acceptance limit, any appropriate means to cool the cavity gas may be employed to reduce the gas temperature to the acceptance criterion. Typically, this may involve intrusive means, such as recirculation cooling of the MPC cavity helium, or non-intrusive means, such as cooling of the exterior surface of the MPC enclosure vessel with water or air. The thermal evaluation should include an evaluation of the cooling process, if required, to determine the appropriate criteria for the cooling process, such as fluid flow rate(s), fluid temperature(s), and the cooling duration required to meet the acceptance criterion. Following fuel cool-down (if required), the MPC is flooded with water. The WRS is positioned for MPC lid-to-shell weld removal. The WRS is then removed with the MPC lid left in place.

The annulus shield is removed and the inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to overpack lifting trunnions. The overpack is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks. The overpack and MPC are returned to the designated preparation area. The annulus water is drained and the MPC and overpack are dispositioned for re-use or waste.

1.2.3 Contents of Package

The HI-STAR 100 packaging is classified as a Type B package under 10CFR71. As the HI-STAR 100 System is designed to transport spent nuclear fuel, the maximum activity of the contents requires that the HI-STAR 100 packaging be classified as Category I in accordance with Regulatory Guide 7.11 [1.2.10]. This section delineates the authorized contents permitted for shipment in the HI-STAR 100 System, including fuel assembly types; non-fuel hardware; neutron sources; physical parameter limits for fuel assemblies and sub-components; enrichment, burnup, cooling time, and decay heat limits; location requirements; and requirements for canning the material.

1.2.3.1 Determination of Design Basis Fuel

The HI-STAR 100 package is designed to transport most types of fuel assemblies generated in the commercial U.S. nuclear industry. Boiling-water reactor (BWR) fuel assemblies have been supplied by General Electric (GE), Siemens (SPC), Exxon Nuclear, ANF, UNC, ABB Combustion Engineering, Allis-Chalmers (AC) and Gulf Atomic. Pressurized-water reactor (PWR) fuel assemblies are generally supplied by Westinghouse, Babcock & Wilcox, ANF, and ABB Combustion Engineering. ANF, Exxon, and Siemens are historically the same manufacturing company under different ownership. Within this report, SPC is used to designate fuel manufactured by ANF, Exxon, or Siemens. Publications such as Refs. [1.2.6], [1.2.7], and [1.2.15] provide a comprehensive description of fuel discharged from U.S. reactors. A central object in the design of the HI-STAR 100 System is to ensure that a majority of SNF discharged from the U.S. reactors can be transported in one of the MPCs.

The cell openings in the fuel basket have been sized to accommodate all BWR and PWR assemblies listed in Refs. [1.2.6], [1.2.7], and [1.2.15], except as noted below. Similarly, the cavity length of the MPC has been set at a dimension that permits transportation of most types of PWR fuel assemblies and BWR fuel assemblies with or without fuel channels. The one exception is as follows:

- The South Texas Units 1 & 2 SNF, and CE 16x16 System 80™ SNF are too long to be accommodated in the available MPC cavity length.

In addition to satisfying the cross sectional and length compatibility, the active fuel region of the SNF must be enveloped in the axial direction by the neutron absorber located in the MPC fuel basket. Alignment of the neutron absorber with the active fuel region is ensured by the use of upper and lower fuel spacers suitably designed to support the bottom and restrain the top of the fuel assembly. The spacers axially position the SNF assembly such that its active fuel region is properly aligned with the neutron absorber in the fuel basket. Figure 1.2.15 provides a pictorial representation of the fuel spacers positioning the fuel assembly active fuel region. Both the upper and lower fuel spacers are designed to perform their function under normal and hypothetical accident conditions of transport. Due to the shorter, custom MPC design for Trojan plant fuel, only lower fuel spacers are needed for certain fuel assemblies that do not contain integral control rod assemblies. This creates the potential for a slight misalignment between the active fuel region of a fuel assembly and the neutron absorber panels affixed to the cell walls of the Trojan MPCs. This condition is addressed in the criticality evaluations described in Chapter 6.

In summary, the geometric compatibility of the SNF with the MPC designs does not require the definition of a design basis fuel assembly. This, however, is not the case for structural, containment, shielding, thermal-hydraulic, and criticality criteria. In fact, the same fuel type in a category (PWR or BWR) may not control the cask design in all of the above-mentioned criteria. To ensure that no SNF listed in Refs. [1.2.6], [1.2.7], and [1.2.15] that is geometrically admissible in the HI-STAR MPC is precluded from loading, it is necessary to determine the governing fuel specification for each analysis criteria. To make the necessary determinations, potential candidate fuel assemblies for each qualification criteria were considered. Table 1.2.8 lists the PWR fuel assemblies evaluated. These fuel assemblies were evaluated to define the governing design criteria for PWR fuel. The BWR fuel assembly designs evaluated are listed in Table 1.2.9. Tables 1.2.10 and 1.2.11 provide the fuel characteristics determined to be acceptable for transport in the HI-STAR 100 System. Each "array/class" listed in these tables represents a bounding set of parameters for one or more fuel assembly types. The array/classes are defined in SAR Section 6.2. Table 1.2.12 lists the BWR and PWR fuel assembly designs that are found to govern for the qualification criteria, namely reactivity, shielding, and thermal. Thermal is broken down into three criteria, namely: 1) fuel assembly effective planar conductivity, 2) fuel basket effective axial conductivity, and 3) MPC density and heat capacity. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Tables 1.2.10, 1.2.11, and 1.2.21 through 1.2.36 provide the specific limits for all material authorized to be transported in the HI-STAR 100 System. Additional information on the design basis fuel definition is presented in the following subsections.

1.2.3.2 Design Payload for Intact Fuel

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for intact fuel to be transported in the HI-STAR 100 System is provided in Tables 1.2.10, 1.2.11, and 1.2.22 through 1.2.36. The placement of a single stainless steel clad fuel assembly in an MPC necessitates that all fuel assemblies (stainless steel clad or Zircaloy clad) stored in that MPC meet the maximum heat generation requirements for stainless steel clad fuel. Stainless steel clad fuel assemblies are not authorized for transportation in the MPC-68F or MPC-32.

Fuel assemblies without fuel rods in fuel rod locations cannot be classified as intact fuel unless dummy fuel rods, which occupy a volume equal to or greater than the original fuel rods, replace the missing rods prior to loading. Any intact fuel assembly that falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly can be safely transported in the HI-STAR 100 System.

The fuel characteristics specified in Tables 1.2.10, 1.2.11, and 1.2.21 have been evaluated in this SAR and are acceptable for transport in the HI-STAR 100 System.

1.2.3.3 Design Payload for Damaged Fuel and Fuel Debris

Damaged fuel and fuel debris are defined in Table 1.0.1. The only PWR damaged fuel and fuel debris authorized for transportation in the HI-STAR 100 System is that from the Trojan plant. The only BWR damaged fuel and fuel debris authorized for transportation in the HI-STAR 100 System is that from the Dresden Unit 1 and Humboldt Bay plants.

Damaged fuel may only be transported in the MPC-24E, MPC-24EF, MPC-68, or MPC-68F as shown in Tables 1.2.23 through 1.2.26. Fuel debris may only be transported in the MPC-24EF and the MPC-68F as shown in Tables 1.2.24 and 1.2.26. Damaged fuel and fuel debris must be transported in stainless steel Holtec damaged fuel containers (DFCs) or other approved stainless steel damaged/failed fuel canister in the HI-STAR 100 System. The list of approved damaged/failed fuel canisters and associated SAR figures are provided below:

- Holtec-designed Dresden Unit 1 and Humboldt Bay Damaged Fuel Container (Figure 1.2.10)
- Sierra Nuclear-designed Trojan Failed Fuel Can (Figure 1.2.10A) containing Trojan damaged fuel, fuel debris, or Trojan Fuel debris process cans; or containing Trojan Fuel Debris Process Can Capsules (Figure 1.2.10C), which themselves contain Trojan Fuel Debris Process Cans (Figure 1.2.10B).
- Holtec-designed Damaged Fuel Container for Trojan plan fuel (Figure 1.2.10D)
- Dresden Unit 1's TN Damaged Fuel Container (Figure 1.2.11)
- Dresden Unit 1's Thoria Rod Canister (Figure 1.2.11A)

1.2.3.3.1 BWR Damaged Fuel and Fuel Debris

Dresden Unit 1 (UO_2 fuel rods and MOX fuel rods) and Humboldt Bay fuel arrays (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) are authorized for transportation as damaged fuel in the MPC-68 and damaged fuel or fuel debris in the MPC-68F. No other BWR damaged fuel or fuel debris is authorized for transportation.

The limits for transporting Dresden Unit 1 and Humboldt Bay damaged fuel and fuel debris are given in Table 1.2.23 and 1.2.24. The placement of a single damaged fuel assembly in an MPC-68 or MPC-68F, or a single fuel debris damaged fuel container in an MPC-68F necessitates that all fuel assemblies (intact, damaged, or debris) placed in that MPC meet the maximum heat generation requirements specified in Tables 1.2.23 and 1.2.24.

The fuel characteristics specified in Tables 1.2.11, 1.2.23 and 1.2.24 for Dresden Unit 1 and Humboldt Bay fuel arrays have been evaluated in this SAR and are acceptable for transport as damaged fuel or fuel debris in the HI-STAR 100 System. Because of the long cooling time, small size, and low weight of spent fuel assemblies qualified as damaged fuel or fuel debris, the DFC and its contents are bounded by the structural, thermal, and shielding analyses performed for the intact BWR design basis fuel. Separate criticality analysis of the bounding fuel assembly for the damaged fuel and fuel debris has been performed in Chapter 6.

As Dresden Unit 1 and Humboldt Bay fuel assemblies classified as fuel debris have significant cladding damage, no cladding integrity is assumed. To meet the double containment criteria of 10CFR71.63(b) for plutonium shipments, the MPC-68F provides the secondary containment boundary (separate inner container), while the overpack provides the primary containment boundary.

The fuel characteristics specified in Table 1.2.11 for the Dresden Unit 1 and Humboldt Bay fuel arrays (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) have been evaluated in this SAR and are acceptable for transport as damaged fuel or fuel debris in the HI-STAR 100 System after being placed in a damaged fuel container.

1.2.3.3.2 PWR Damaged Fuel and Fuel Debris

The PWR damaged fuel and fuel debris authorized for transportation in the HI-STAR 100 System is limited to that from the Trojan plant. The limits for transporting Trojan plant damaged fuel and fuel debris in the Trojan MPC-24E/EF are given in Tables 1.2.10, 1.2.25 and 1.2.26. All Trojan plant damaged fuel, and fuel debris listed below is authorized for transportation in the HI-STAR 100 System [1.2.12]:

- Damaged fuel assemblies in Trojan failed fuel cans

- Damaged fuel assemblies in Holtec's Trojan plant PWR damaged fuel container

- Fuel assemblies classified as fuel debris in Trojan failed fuel cans

- Trojan fuel assemblies classified as fuel debris in Holtec's Trojan damaged fuel container

- Fuel debris consisting of loose fuel pellets, fuel pellet fragments, and fuel assembly metal fragments (portions of fuel rods, portions of grid assemblies, bottom nozzles, etc.) in Trojan failed fuel cans

- Trojan fuel debris process cans loaded into Trojan fuel debris process can capsules and then into Trojan failed fuel cans. The fuel debris process cans contain fuel debris (metal fragments) and were used to process organic media removed from the Trojan spent fuel pool during cleanup operations in preparation for decommissioning the pool. The fuel debris process cans have metallic filters in the can bottom and lid that allowed removal of water and organic media using high temperature steam, while retaining the solid residue from the processed media and fuel debris inside the process can†. Up to five process cans can be loaded into a process can capsule, which is vacuumed, purged, backfilled with helium, and seal-welded closed to provide a sealed containment for the fuel debris.

One Trojan Failed Fuel Can is not completely filled with fuel debris. Therefore, a stainless steel failed fuel can spacer is installed in this FFC to minimize movement of the fuel debris during normal transportation and hypothetical accident conditions. The spacer is a long, square tube with a baseplate that rests atop the fuel debris inside the Trojan FFC. A drawing of the Trojan failed fuel can spacer is provided in Section 1.4. A summary of the structural analysis of the FFC spacer is provided in Section 2.6.1.3.1.3.

1.2.3.4 Structural Payload Parameters

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are listed in Tables 1.2.22 through 1.2.27 for the various MPC models. The centers of gravity reported in Chapter 2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower spacers are designed to withstand normal and accident conditions of transport. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 1.2.16 and 1.2.17. Due to the custom design of the Trojan MPCs, only lower fuel spacers are required with Trojan plant fuel assemblies not containing non-fuel hardware or neutron sources. In order to qualify for transport in the HI-STAR 100 MPC, the SNF must satisfy the physical parameters listed in Tables 1.2.21 through 1.2.36, as applicable.

1.2.3.5 Thermal Payload Parameters

The principal thermal design parameter for the fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly and the decay heat removal capabilities of the HI-STAR 100 System. The maximum heat generation rate per assembly for the design basis fuel assembly is based on the fuel assembly type with the lowest thermal performance characteristics. The parameters that define this decay heat design basis fuel are listed in Table 1.2.12. The governing thermal parameters to ensure that the range of SNF discussed previously are bounded by the thermal analysis discussed in detail and specified in Chapter 3. By utilizing these bounding thermal parameters, the calculated peak fuel rod cladding temperatures are conservative for the actual spent fuel assemblies, which are apt to have a higher thermal conductivity.

† The Trojan Fuel Debris Process Cans were used in the spent fuel pool cleanup effort conducted as part of plant decommissioning. This project is complete and not associated with certification of Trojan fuel debris for transportation in the HI-STAR 100 System under 10 CFR 71.

The peak fuel cladding temperature limit for normal conditions of transport is 400°C (752°F), which is consistent with the guidance in ISG-11, Revision 3~~2~~ [1.2.14]. Tables 1.2.21 through 1.2.27 provide the maximum heat generation for all fuel assemblies authorized for transportation in the HI-STAR 100 System. The basis for these limits is discussed in Chapter 3.

Finally, the axial variation in the heat emission rate in the design basis fuel is defined based on the axial burnup distribution. For this purpose, the data provided in Refs. [1.2.8], [1.2.9], and [1.2.12] are utilized and summarized in Table 1.2.15 and Figures 1.2.13, 1.2.13A, and 1.2.14, for reference. These distributions are representative of fuel assemblies with the design burnup levels considered. These distributions are used for analysis only, and do not provide a criteria for fuel assembly acceptability for transport in the HI-STAR 100 System.

1.2.3.6 Radiological Payload Parameters

The principal radiological design criteria are the 10CFR71.47 and 10CFR71.51 radiation dose rate and release requirements for the HI-STAR 100 System. The radiation dose rate is directly affected by the gamma and neutron source terms of the SNF assembly.

The gamma and neutron sources are separate and are affected differently by enrichment, burnup, and cool time. It is recognized that, at a given burnup, the radiological source terms increase monotonically as the initial enrichment is reduced. The shielding design basis fuel assembly is, therefore, evaluated for different combinations of maximum burnup, minimum cooling time, and minimum enrichment. The shielding design basis intact fuel assembly thus bounds all other intact fuel assemblies.

The design basis dose rates can be met by a variety of burnup levels, cooling times, and minimum enrichments. Tables 1.2.21 through 1.2.36 include the burnup and cooling time values that meet the radiological dose rate requirements for all authorized contents to be transported in each MPC model. The allowable maximum burnup, minimum cooling time, and minimum enrichment limits were chosen strictly based on the dose rate requirements. All allowable burnup, cooling time, and minimum enrichment combinations result in calculated dose rates less than the regulatory dose rate limits.

Table 1.2.15 and Figures 1.2.13. 1.2.13A, and 1.2.14 provide the axial distribution for the radiological source term for PWR and BWR fuel assemblies, and for Trojan plant-specific fuel, based on the actual burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analysis only, and do not provide criteria for fuel assembly acceptability for transport in the HI-STAR 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 1.2.21 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for transport. Up to one Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel, up to a total of 68 may be transported.

1.2.3.7 Criticality Payload Parameters

As discussed earlier, the MPC-68/68F and MPC-32 feature a basket without flux traps. In these fuel baskets, there is one panel of neutron absorber between adjacent fuel assemblies. The MPC-24/24E/24EF employs a construction wherein two neighboring fuel assemblies are separated by two panels of neutron absorber with a water gap between them (flux trap construction). The MPC-24 flux trap basket can accept a much higher enrichment fuel than a non-flux trap basket without taking credit for fuel assembly burnup in the criticality analysis. The maximum initial ^{235}U enrichment for PWR and BWR fuel authorized for transport is specified by fuel array/class in Tables 1.2.10 and 1.2.11, respectively. Trojan plant fuel is limited to a lower maximum initial enrichment of 3.7 wt.% ^{235}U compared to other fuel in its array/class, based on the specific analysis performed for the custom-designed Trojan MPCs containing only Trojan plant fuel.

The MPC-24 Boral ^{10}B areal density is specified at a minimum loading of 0.0267 g/cm^2. The MPC-24E/EF, MPC-32, and MPC-68 Boral ^{10}B areal density is specified at a minimum loading of 0.0372 g/cm^2. The MPC-68F Boral ^{10}B areal density is specified at a minimum loading of 0.01 g/cm^2.

For all MPCs, the ^{10}B loading areal density used for analysis is conservatively established at 75% of the minimum ^{10}B areal density to demonstrate that the reactivity under the most adverse accumulation of tolerances and biases is less than 0.95. The reduction in ^{10}B areal density credit meets NUREG-1617 [1.0.5], which requires a 25% reduction in ^{10}B areal density credit. A large body of sampling data accumulated by Holtec from thousands of manufactured Boral panels indicates the average ^{10}B areal densities to be approximately 15% greater than the specified minimum.

Credit for burnup of the fuel, in accordance with the intent of the guidance in Interim Staff Guidance Document 8 (ISG-8) [1.2.13], is taken in the criticality analysis to allow the transportation of certain PWR fuel assemblies in MPC-32. Burnup credit is a required input to qualify PWR fuel for transportation in the MPC-32, considering the inleakage of moderator (i.e., unborated water) under accident conditions. This hypothetical event is non-credible given the double barrier design engineered into the HI-STAR 100 System with the fully welded MPC enclosure vessel (designed for 60 g's) surrounded by the sealed overpack, which is designed for deep submersion under water (greater than 650 feet submersion) without breach. The details of the burnup credit analyses are provided in Chapter 6, including detailed discussion of how the recommendations of ISG-8 were implemented. Exceptions to some of the recommendations in ISG-8 were necessary (e.g., partial credit for fission products) in order to develop burnup versus enrichment curves that can be practically implemented at the plants. These exceptions are described in Chapter 6.

1.2.3.8 Non-Fuel Hardware and Neutron Sources

BWR fuel is permitted to be stored with or without Zircaloy channels. Control blades and stainless steel channels are not authorized for transportation in the HI-STAR 100 System. Dresden Unit 1 (D-1) neutron sources are authorized for transportation a shown in Tables 1.2.23 and 1.2.24. The D-1 neutron sources are single, long rods containing Sb-Be source material that fits into a water rod location in a D-1 fuel assembly.

Except for Trojan plant fuel, no PWR non-fuel hardware or neutron sources are authorized for transportation in the HI-STAR 100 System. For Trojan plant fuel only, the following non-fuel hardware and neutron sources are permitted for transportation in specific quantities as shown in Tables 1.2.25 and 1.2.26:

- Rod Cluster Control Assemblies (RCCAs) with cladding made of Type 304 stainless steel and Ag-In-Cd neutron absorber material.
- Burnable Poison Rod Assemblies (BPRAs) with cladding made of Type 304 stainless steel and borosilicate glass tube neutron poison material.
- Thimble Plug Devices made of Type 304 stainless steel.
- Neutron source assemblies with cladding made of Type 304 stainless steel - two (2) californium primary source assemblies and four (4) antimony-beryllium secondary source assemblies.

These devices are designed with thin rods of varying length and materials as discussed above, that fit into the fuel assembly guide tubes within the fuel rod lattice. The upper fittings for each device can vary to accommodate the handling tool (grapple) design. During reactor operation, the positions of the RCCAs are controlled by the operator using the control rod drive system, while the BPRAs, TPDs, and neutron sources stay fully inserted.

A complete list of the authorized non-fuel hardware and neutron sources, including appropriate limits on the characteristics of this material, is provided in Tables 1.2.23 through 1.2.36, as applicable.

1.2.3.9 Summary of Authorized Contents

The criticality safety index for the HI-STAR 100 Package is zero. A fuel assembly is acceptable for transport in a HI-STAR 100 System if it fulfills the following criteria.

a. It satisfies the physical parameter characteristics listed in Tables 1.2.10 or 1.2.11, as applicable..

b. It satisfies the cooling time, decay heat, burnup, enrichment, and other limits specified in Tables 1.2.21 through 1.2.36, as applicable.

c. Deleted.

d. Deleted.

A damaged fuel assembly shall be transported in a damaged fuel container or other authorized damaged/failed fuel canister, and shall meet the characteristics specified in Tables 1.2.23 through 1.2.26 for transport in the MPC-68, MPC-68F, MPC-24E, or MPC-24EF. Fuel classified as fuel debris shall be placed in a damaged fuel container or other authorized damaged/failed fuel canister and shall meet the characteristics specified in Tables 1.2.24 or 1.2.26 for transport in the MPC-68F or MPC-24EF.

Stainless steel clad fuel assemblies shall meet the characteristics specified in Tables 1.2.22 through 1.2.33 for transport in the MPC-24, MPC-24E, MPC-24EF, or MPC-68.

MOX BWR fuel assemblies shall meet the requirements of Tables 1.2.23 or 1.2.24 for intact and damaged fuel/fuel debris.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 1.2.21 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for transport. Up to one Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel, up to a total of 68 may be transported.

Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68 or MPC-68F.

Table 1.2.2 summarizes the key system data for the HI-STAR 100 System. Table 1.2.3 summarizes the key parameters and limits for the HI-STAR 100 MPCs. Tables 1.2.10, 1.2.11, and 1.2.21 through 1.2.37 and other tables referenced from these tables provide the limiting conditions for all material to be transported in the HI-STAR 100 System.

Table 1.2.1

TABLE INTENTIONALLY DELETED

Table 1.2.2

SUMMARY OF KEY SYSTEM DATA FOR HI-STAR 100

PARAMETER	VALUE (Nominal)	
Types of MPCs in this SAR	6	4 for PWR 2 for BWR
MPC capacity	MPC-24	Up to 24 intact ZR or stainless steel clad PWR fuel assemblies
	MPC-24E	Up to 24 intact ZR or stainless steel clad PWR fuel assemblies. Up to four (4) Trojan plant fuel assemblies classified as damaged fuel, each in a Trojan Failed Fuel Can or a Holtec damaged fuel container, and the complement intact fuel assemblies.
	MPC-24EF	Up to 24 intact ZR or stainless steel clad PWR fuel assemblies. Up to four (4) Trojan plant fuel assemblies classified as damaged fuel or fuel debris, each in a Trojan Failed Fuel Can or a Holtec damaged fuel container; or other Trojan fuel debris stored in Trojan Process Cans either placed directly into a Trojan Failed Fuel Can or placed inside Trojan Process Can Capsules and then in Trojan Failed Fuel Cans; and the complement intact fuel assemblies.
	MPC-32	Up to 32 intact ZR-clad PWR fuel assemblies.
	MPC-68	Up to 68 intact ZR or stainless steel clad BWR fuel assemblies or damaged ZR clad fuel assemblies* in damaged fuel containers within an MPC-68
	MPC-68F	Up to 4 damaged fuel containers with ZR clad BWR fuel debris* and the complement intact or damaged* ZR clad BWR fuel assemblies within an MPC-68F. *Only damaged fuel and fuel debris from Dresden Unit 1 or Humboldt Bay is authorized for transportation in the MPC-68 and MPC-68F.

Table 1.2.3
KEY PARAMETERS FOR HI-STAR 100 MULTI-PURPOSE CANISTERS

PARAMETER	PWR	BWR
Unloaded MPC weight (lb)	See Table 2.2.1	See Table 2.2.1
Minimum neutron absorber ^{10}B loading (g/cm^2)	0.0267 (MPC-24) 0.0372 (MPC-24E/EF) 0.0372 (MPC-32)	0.0372 (MPC-68) 0.01 (MPC-68F)
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725°†/-40°††	725°†/-40°††
Design Internal pressure (psig) Normal Conditions Off-normal Conditions Accident Conditions	 100 100 200	 100 100 200
Total heat load, max. (kW)	20.0	18.5
Maximum permissible peak fuel cladding temperature (°F)	752° (normal conditions) 1058° (accident conditions)	752° (normal conditions) 1058° (accident conditions)
MPC internal environment Helium filled (psig)	$\geq$ 0 and $\leq$ 44.8 psig††† at a reference temperature of 70°F	$\geq$ 0 and $\leq$ 44.8 psig††† at a reference temperature of 70°F
MPC external environment/overpack internal environment Helium filled initial pressure (psig, at STP)	$\geq$ 10 and $\leq$ 14	$\geq$ 10 and $\leq$ 14
Maximum permissible reactivity including all uncertainty and biases	<0.95	<0.95
End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

† Maximum normal condition design temperature for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.1.2

†† Temperature based on minimum ambient temperature (10CFR71.71(c)(2)) and no fuel decay heat load.

††† This value represents the nominal backfill value used in the thermal analysis, plus 2 psig operating tolerance. Based on the MPC pressure results in Table 3.4.15 and the pressure limits specified in Table 2.1.1, there is sufficient analysis margin to accommodate this operating tolerance.

Tables 1.2.4 through 1.2.6

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Table 1.2.7

HI-STAR 100 LOADING OPERATIONS DESCRIPTION

Site-specific handling and operating procedures will be prepared, reviewed, and approved by each owner/user.	
1	Overpack and MPC lowered into the fuel pool without closure plate and MPC lid
2	Fuel assemblies transferred to the MPC fuel basket
3	MPC lid lowered onto the MPC
4	Overpack/MPC assembly moved to the decon pit and MPC lid welded in place, examined, pressure tested, and leak tested
5	MPC dewatered, dried, backfilled with helium, and the vent/drain port cover plates and closure ring welded
6	Overpack drained and external surfaces decontaminated
7	Overpack seals and closure plate installed and bolts pre-tensioned
8	Overpack cavity dried, backfilled with helium, and helium leak tested
9	HI-STAR 100 System transferred to transport bay
10	HI-STAR 100 placed onto transport saddles, tied down, impact limiters and personnel barrier installed, and package surveyed for release for transport.

Table 1.2.8

PWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	**Array Type**
B&W 15x15	All
B&W 17x17	All
CE 14x14	All
CE 16x16	All except System 80™
WE 14x14	All
WE 15x15	All
WE 17x17	All
St. Lucie	All
Ft. Calhoun	All
Haddam Neck (Stainless Steel Clad)	All
San Onofre 1 (Stainless Steel Clad, except MOX)	All
Indian Point 1	All

Table 1.2.9

BWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	Array Type			
GE BWR/2-3	All 7x7	All 8x8	All 9x9	All 10x10
GE BWR/4-6	All 7x7	All 8x8	All 9x9	All 10x10
Humboldt Bay	All 6x6	All 7x7 (Zircaloy Clad)		
Dresden-1	All 6x6	All 8x8		
LaCrosse (Stainless Steel Clad)	All			

Table 1.2.10
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 5.0 (24E/24EF)	≤ 5.0
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	N/A	N/A
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table 1.2.10 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	*(Note 5)*≤ ~~5.0~~	*(Note 5)*≤ ~~5.0~~	*(Note 5)*≤ ~~5.0~~
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 1.2.10 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	*(Note 5)*≤ ~~5.0~~	N/A	*(Note 5)*≤ ~~5.0~~	*(Note 5)*≤ ~~5.0~~	*(Note 5)*≤ ~~5.0~~
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≥ 0.563	≥ 0.568	≥ 0.506	≥ 0.496	≥ 0.496	≥ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 1.2.10 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

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

2. ZR designates any zirconium-based fuel cladding material authorized for use in a commercial power reactor.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances.

4. Each guide tube replaces four fuel rods.

5. *"N/A" means that this array/class is not authorized for transportation in the MPC-32. For authorized array/classes,* ~~M~~*m*inimum assembly average burnup *and maximum enrichment* is ~~required~~*specified in*~~-per~~ Table 1.2.34.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table 1.2.11
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods.	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 1.2.11 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table 1.2.11 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9 B	9x9 C	9x9 D	9x9 E (Note 13)	9x9 F (Note 13)	9x9 G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table 1.2.11 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10 A	10x10 B	10x10 C	10x10 D	10x10 E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 1.2.11 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

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

2. ZR designates any zirconium-based fuel cladding material authorized for use in a commercial power reactor.

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

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable.

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

9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.

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

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

12. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.

Table 1.2.12

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	MPC-68/68F	MPC-24/24E/24EF/32
Reactivity	SPC 9x9-5 (Array/Class 9x9E/F)	B&W 15x15 (Array/Class 15x15F)
Shielding (Source Term)	GE 7x7	B&W 15x15
Fuel Assembly Effective Planar Thermal Conductivity	GE 11 9x9	W 17x17 OFA
Fuel Basket Effective Axial Thermal Conductivity	GE 7x7	W 14x14 OFA
MPC Density and heat Capacity	GE 7x7	W 14x14 OFA

Tables 1.2.13 and 1.2.14

INTENTIONALY DELETED

Table 1.2.15

NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

GENERIC FUEL DISTRIBUTION†			
Interval	Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)	PWR Fuel Normalized Distribution	BWR Fuel Normalized Distribution
1	0% to 4-1/6%	0.5485	0.2200
2	4-1/6% to 8-1/3%	0.8477	0.7600
3	8-1/3% to 16-2/3%	1.0770	1.0350
4	16-2/3% to 33-1/3%	1.1050	1.1675
5	33-1/3% to 50%	1.0980	1.1950
6	50% to 66-2/3%	1.0790	1.1625
7	66-2/3% to 83-1/3%	1.0501	1.0725
8	83-1/3% to 91-2/3%	0.9604	0.8650
9	91-2/3% to 95-5/6%	0.7338	0.6200
10	95-5/6% to 100%	0.4670	0.2200

TROJAN PLANT FUEL DISTRIBUTION††		
Interval	Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)	Normalized Distribution
1	0% to 5%	0.59
2	5% to 10%	0.89
3	10% to 15%	1.03
4	15% to 20%	1.07
5	20% to 25%	1.09
6	25% to 45%	1.10
7	45% to 70%	1.09
8	70% to 75%	1.07
9	75% to 80%	1.05
10	80% to 85%	1.02
11	85% to 90%	0.96
12	90% to 95 %	0.82
13	95% to 100%	0.56

† References [1.2.8] and [1.2.9]

†† Reference [1.2.12]

Table 1.2.16

SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS (Note 1)

Fuel Assembly Type	Assembly Length w/o NFH† (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
CE 14x14	157	4.1	137	9.5	10
CE 16x16	176.8	4.7	150	0	0
BW 15x15	165.7	8.4	141.8	6.7	4.1
W 17x17 OFA	159.8	3.7	144	8.2	8.5
W 17x17S	159.8	3.7	144	8.2	8.5
W 17x17V5H	160.1	3.7	144	7.9	8.5
W 15x15	159.8	3.7	144	8.2	8.5
W 14x14S	159.8	3.7	145.2	9.2	7.5
W 14x14 OFA	159.8	3.7	144	8.2	8.5
Ft. Calhoun	146	6.6	128	10.25	20.25
St. Lucie 2	158.2	5.2	136.7	10.25	8.05
B&W 15x15 SS	137.1	3.873	120.5	19.25	19.25
W 15x15 SS	137.1	3.7	122	19.25	19.25
W 14x14 SS	137.1	3.7	120	19.25	19.25
Indian Point 1	137.2	17.705	101.5	18.75	20.0

Notes: 1. These fuel spacer lengths are not applicable to Trojan plant fuel. Trojan plant fuel spacer lengths are determined uniquely for the custom-designed Trojan MPC-24E/EF, as necessary, based on the presence of non-fuel hardware. They are sized to maintain the active fuel within the envelope of the neutron absorber affixed to the cell walls and allow for an approximate 2-inch gap between the fuel and the MPC lid. See Chapter 6 for discussion of potential misalignments between the active fuel and the neutron absorber.

† NFH is an abbreviation for non-fuel hardware, including control components. Fuel assemblies with control components may require shorter fuel spacers.

Table 1.2.17

SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS (Note 1)

Fuel Assembly Type	Assembly Length (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
GE/2-3	171.2	7.3	150	4.8	0
GE/4-6	176.2	7.3	150	0	0
Dresden 1	134.4	11.2	110	18	28.0
Humboldt Bay	95	8	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	142.1†	11.2	110	17	16.9
Humboldt Bay Damaged Fuel or Fuel Debris	105.5†	8	79	35.25	35.25
LaCrosse	102.5	10.5	83	37	37.5

Notes: 1. Each user shall specify the fuel spacer lengths based on their fuel length and allowing an approximate 2-inch gap between the fuel and the MPC lid. See Chapter 6 for discussion of potential misalignments between the active fuel and the neutron absorber.

† Fuel length includes the damaged fuel container.

Aspect of Post-Accident Performance	Results with Demonstrated Integrity of MPC Enclosure Vessel	Results with Postulated Gross Failure of MPC Enclosure Vessel
Containment Boundary Integrity	The MPC enclosure vessel is leak tested to 5.0×10^{-6} atm cm^3/s (helium). The overpack containment boundary is standard air leak tested to 4.3×10^{-6} atm cm^3/s (helium). Both boundaries are shown to withstand all hypothetical accident conditions. Therefore, there will be no detectable release of radioactive materials.	The overpack containment boundary is leak tested to 4.3×10^{-6} atm cm^3/s (helium). The overpack containment boundary is shown to withstand all hypothetical accident conditions. Therefore, the overpack containment boundary meets the accident condition leakage rates.
Maintenance of Subcritical Margins (Maximum k_{eff})	The MPC enclosure vessel is seal welded and there is no breach of the MPC. The bolted closure overpack containment boundary has been shown to prevent water immersion. Therefore, the maximum reactivity of the fuel in a dry MPC is less than 0.5.	The bolted closure overpack containment boundary has been shown to prevent water immersion. Therefore, the maximum reactivity of the fuel in a dry MPC is less than 0.5. Assuming the MPC is fully flooded with water, the reactivity is shown to be below the regulatory requirement of 0.95 including uncertainties and bias.
Adequate Shielding	The MPC enclosure vessel boundary has no effect on the dose rates of the HI-STAR 100 System.	Failure of the MPC enclosure vessel to maintain a release boundary has no effect on the dose rates of the HI-STAR 100 System.
Adequate Heat Rejection (Peak Fuel Cladding Temperature)	The MPC enclosure vessel maintains the helium and the peak fuel cladding temperature is demonstrated to remain below 800°F in the post-fire hypothetical accident condition.	Assuming the MPC internal helium fill pressure is released into the overpack containment, the pressure within the small annulus would rise to equalize with the MPC internal pressure. There would be a corresponding slight pressure decrease in the MPC enclosure vessel. The comparatively small volume of the annulus and pressure differential results in the slight pressure change. This will have a negligibly small effect on the peak fuel cladding temperature. The overpack containment boundary is demonstrated to withstand all hypothetical accident conditions. Therefore, there is no credible mechanism for the release of the helium.

Tables 1.2.19 and 1.2.20

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Table 1.2.21

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

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

Table 1.2.22

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 1.2.10 for the applicable array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 1.2.10 for the applicable array/class
Maximum Initial Enrichment	As specified in Table 1.2.10 for the applicable array/class
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	ZR clad: As specified in Table 1.2.28 or Table 1.2.29, as applicable SS clad: As specified in Table 1.2.30
Decay Heat Per Assembly	ZR clad: ≤ 833 Watts SS clad: ≤ 488 Watts
Fuel Assembly Length	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,680 lbs
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies. ▪ Non-fuel hardware and neutron sources not permitted. ▪ Damaged fuel assemblies and fuel debris not permitted. ▪ Trojan plant fuel not permitted.

Table 1.2.23

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-68

PARAMETER	VALUE (Note 1)			
Fuel Type(s)	Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 1.2.11 for the applicable array/class, with or without Zircaloy channels	Uranium oxide, BWR damaged fuel assemblies meeting the limits in Table 1.2.11 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels, placed in Damaged Fuel Containers(DFCs)	Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 1.2.11 for array/class 6x6B, with or without Zircaloy channels	Mixed Oxide (MOX) BWR damaged fuel assemblies meeting the limits in Table 1.2.11 for array/class 6x6B, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 1.2.11 for the applicable array/class	ZR	ZR	ZR
Maximum Initial Planar-Average and Rod Enrichment	As specified in Table 1.2.11 for the applicable array/class	As specified in Table 1.2.11 for the applicable array/class	As specified in Table 1.2.11 for array/class 6x6B	As specified in Table 1.2.11 for array/class 6x6B
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	ZR clad: As specified in Table 1.2.31 except as provided in Notes 2 and 3 SS clad: Note 4	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTU, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTIHM, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTIHM, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.
Decay Heat Per Assembly	ZR clad: $\leq$ 272 Watts (Note 5) SS clad: $\leq$ 83 Watts	$\leq$ 115 Watts	$\leq$ 115 Watts	$\leq$ 115 Watts

Table 1.2.23 (cont'd)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-68

PARAMETER	VALUE (Note 1)			
Fuel Assembly Length	≤ 176.2 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)
Fuel Assembly Width	≤ 5.85 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)
Fuel Assembly Weight	≤ 700 lbs (including channels)	≤ 550 lbs, (including channels and DFC)	≤ 400 lbs, (including channels)	≤ 550 lbs, (including channels and DFC)
Quantity per MPC	Up to 68 BWR intact fuel assemblies	Up to 68 BWR damaged and/or intact fuel assemblies	Up to 68 BWR intact fuel assemblies	Up to 68 BWR damaged and/or intact fuel assemblies
Other Limitations	▪ Quantity is limited to up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 1.2.21 plus any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in DFCs and intact fuel assemblies up to a total of 68. ▪ Stainless steel channels are not permitted. ▪ Fuel debris is not permitted. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.			

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for transportation.

2. Array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt. % ^{235}U.

3. Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt. % ^{235}U.

4. SS-clad fuel assemblies shall have a cooling time ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt. % ^{235}U.

5. Array/class 8x8F fuel assemblies shall have a decay heat ≤ 183.5 Watts.

Table 1.2.24

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-68F

PARAMETER	VALUE (Notes 1 and 2)			
Fuel Type(s)	Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 1.2.11 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels	Uranium oxide, BWR damaged fuel assemblies or fuel debris meeting the limits in Table 1.2.11 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels, placed in Damaged Fuel Containers(DFCs)	Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 1.2.11 for array/class 6x6B, with or without Zircaloy channels	Mixed Oxide (MOX) BWR damaged fuel assemblies or fuel debris meeting the limits in Table 1.2.11 for array/class 6x6B, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs))
Cladding Type	ZR	ZR	ZR	ZR
Maximum Initial Planar-Average and Rod Enrichment	As specified in Table 1.2.11 for the applicable array/class	As specified in Table 1.2.11 for the applicable array/class	As specified in Table 1.2.11 for array/class 6x6B	As specified in Table 1.2.11 for array/class 6x6B
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTU, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTU, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTIHM, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.	Cooling time $\geq$ 18 years, average burnup $\leq$ 30,000 MWD/MTIHM, and minimum initial enrichment $\geq$ 1.8 wt. % ^{235}U.
Decay Heat Per Assembly	$\leq$ 115 Watts	$\leq$ 115 Watts	$\leq$ 115 Watts	$\leq$ 115 Watts

Table 1.2.24 (cont'd)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-68F

PARAMETER	VALUE (Note 1)			
Fuel Assembly Length	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)
Fuel Assembly Width	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)
Fuel Assembly Weight	≤ 400 lbs (including channels)	≤ 550 lbs (including channels and DFC)	≤ 400 lbs (including channels)	≤ 550 lbs (including channels and DFC)
Other Limitations	▪ Quantity is limited to up to four (4) DFCs containing Dresden Unit 1 or Humboldt Bay uranium oxide or MOX fuel debris. The remaining fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable: - uranium oxide BWR intact fuel assemblies - MOX BWR intact fuel assemblies - uranium oxide BWR damaged fuel assemblies in DFCs - MOX BWR damaged fuel assemblies in DFCs - up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 1.2.21 ▪ Stainless steel channels are not permitted. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.			

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for transportation.

2. Only fuel from Dresden Unit 1 and Humboldt Bay plant are permitted for transportation in the MPC-68F.

Table 1.2.25

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24E

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 1.2.10 for the applicable array/class	Trojan plant damaged fuel meeting the limits in Table 1.2.10 for array/class 17x17B, placed in a Holtec Damaged Fuel Container (DFC) designed for Trojan plan fuel or a Trojan Failed Fuel Can (FFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 1.2.10 for the applicable array/class	ZR
Maximum Initial Enrichment	As specified in Table 1.2.10 for the applicable array/class	3.7 wt. % ^{235}U
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly (except Trojan plant fuel and non-fuel hardware)	ZR clad: As specified in Table 1.2 28 or 1.2.29, as applicable SS clad: As specified in Table 1.2.30	Not applicable
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly for Trojan plant fuel	As specified in Table 1.2.35	As specified in Table 1.2.35
Post-irradiation Cooling Time and Burnup for Trojan plant Non-fuel Hardware and Neutron Sources	As specified in Table 1.2.36	Not applicable
Decay Heat Per Assembly (except for Trojan plant fuel)	ZR clad: $\leq$ 833 Watts SS clad: $\leq$ 488 Watts	Not applicable
Decay heat per Assembly for Trojan plant fuel	$\leq$ 725 Watts	$\leq$ 725 Watts

Table 1.2.25 (cont'd)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24E

PARAMETER	VALUE (Note 1)	
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 169.3 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.43 in. (nominal design)
Fuel Assembly Weight	≤ 1680 lbs (including non-fuel hardware)	≤ 1680 lbs (including DFC or Failed Fuel Can)
Other Limitations	▪ Quantity per MPC: up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with Trojan plant intact fuel assemblies. ▪ Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed (see Figure 1.1.5). Fuel from other plants is not permitted to be transported in the Trojan MPCs. ▪ Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location. ▪ Trojan plant damaged fuel assemblies must be transported in a Holtec DFC for Trojan plant fuel or a Trojan plant FFC. ▪ One (1) Trojan plant Sb-Be and/or two (2) Cf neutron sources, each in a Trojan plant intact fuel assembly may be transported in any one MPC. Each neutron source may be transported in any fuel storage location. ▪ Fuel debris is not authorized for transportation in the MPC-24E. ▪ Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location with damaged fuel assemblies.	

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for transportation.

Table 1.2.26

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24EF

PARAMETER	VALUE (Note 1)		
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 1.2.10 for the applicable array/class	Trojan plant damaged fuel meeting the limits in Table 1.2.10 for array/class 17x17B, placed in a Holtec Damaged Fuel Container (DFC)designed for Trojan plant fuel or a Trojan Failed Fuel Can (FFC)	Trojan plant Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria in Table 1.2.10 for array/class 17x17B, placed in a Holtec Damaged Fuel Container (DFC) designed for Trojan plant fuel or a Trojan Failed Fuel Can (FFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 1.2.10 for the applicable array/class	ZR	ZR
Maximum Initial Enrichment	As specified in Table 1.2.10 for the applicable array/class	$\leq$ 3.7 wt. % ^{235}U	$\leq$ 3.7 wt. % ^{235}U
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly (except Trojan plant fuel and non-fuel hardware)	ZR clad: As specified in Table 1.2 28 or 1.2.29, as applicable SS clad: As specified in Table 1.2.30	Not applicable	Not applicable
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly for Trojan plant fuel	As specified in Table 1.2.35	As specified in Table 1.2.35	As specified in Table 1.2.35

Table 1.2.26 (cont'd)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24EF

PARAMETER	VALUE (Note 1)		
Post-irradiation Cooling Time and Burnup for Trojan plant Non-fuel Hardware and Neutron Sources	As specified in Table 1.2.36	As specified in Table 1.2.36	As specified in Table 1.2.36
Decay Heat Per Assembly (except for Trojan plant fuel)	ZR clad: ≤ 833 Watts SS clad: ≤ 488 Watts	Not applicable	Not applicable
Decay heat per Assembly for Trojan plant fuel	≤ 725 Watts	≤ 725 Watts	≤ 725 Watts
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 169.3 in. (nominal design)	≤ 169.3 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.43 in. (nominal design)	≤ 8.43 in. (nominal design)
Fuel Assembly Weight	≤ 1680 lbs (including non-fuel hardware(	≤ 1680 lbs (including DFC or Failed Fuel Can)	≤ 1680 lbs (including DFC or Failed Fuel Can)

Table 1.2.26 (cont'd)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-24EF

Other Limitations	▪ Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with Trojan plant intact fuel assemblies. ▪ Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed (see Figure 1.1.5). Fuel from other plants is not permitted to be transported in the Trojan MPCs. ▪ Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location. ▪ Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec DFC for Trojan plant fuel. ▪ One (1) Trojan plant Sb-Be and/or two (2) Cf neutron sources, each in a Trojan plant intact fuel assembly may be transported in any one MPC. Each neutron source may be transported in any fuel storage location.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for transportation.

Table 1.2.27

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-32 ~~(Note 1)~~

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 1.2.10 for array/classes 15x15D, E, F, and H and 17x17A, B, and C
Cladding Type	ZR
Maximum Initial Enrichment	As specified in Table 1.2.10
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	As specified in Table 1.2.32 or Table 1.2.33, as applicable
Decay Heat Per Assembly	≤ 625 Watts
Minimum Burnup per Assembly	As specified in Table 1.2.34 for the applicable array/class
Fuel Assembly Length	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,680 lbs
Operating Parameters During Irradiation of the Assembly	
Average in-core soluble boron concentration	*≤ 1000 ppmb*
Average Core outlet water temperature	*≤ 601 K for array/classes 15x15D, E, F and H* *≤ 610 K for array/classes 17x17A, B and C*
Average Specific Power	*≤ 47.36 kW/kg-U for array/classes 15x15D, E, F and H* *≤ 61.61 kW/kg-U for array/classes 17x17A, B and C*

Table 1.2.27 (continued)

LIMITS FOR MATERIAL TO BE TRANSPORTED IN MPC-32

Other Limitations	▪ Quantity is limited to up to 32 PWR intact fuel assemblies in the above-specified array/classes only. ▪ Non-fuel hardware and neutron sources not permitted. ▪ Damaged fuel assemblies and fuel debris not permitted. ▪ Trojan plant fuel not permitted.

~~NOTES:~~

~~1. The MPC-32 is not authorized for transportation in the HI-STAR 100 System at this time.~~

Table 1.2.28

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-24/24E/24EF; PWR FUEL WITH ZR CLADDING AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
≥ 9	$\leq 24,500$	≥ 2.3
≥ 11	$\leq 29,500$	≥ 2.6
≥ 13	$\leq 34,500$	≥ 2.9
≥ 15	$\leq 39,500$	≥ 3.2
≥ 18	$\leq 44,500$	≥ 3.4

Table 1.2.29

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-24/24E/24EF;PWR FUEL WITH ZR CLADDING AND WITH ZIRCALOY IN-CORE GRID SPACERS

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
$\geq$ 6	$\leq$ 24,500	$\geq$ 2.3
$\geq$ 7	$\leq$ 29,500	$\geq$ 2.6
$\geq$ 9	$\leq$ 34,500	$\geq$ 2.9
$\geq$ 11	$\leq$ 39,500	$\geq$ 3.2
$\geq$ 14	$\leq$ 44,500	$\geq$ 3.4

Table 1.2.30

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-24/24E/24EF; PWR FUEL WITH STAINLESS STEEL CLADDING

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
$\geq$ 19	$\leq$ 30,000	$\geq$ 3.1
$\geq$ 24	$\leq$ 40,000	$\geq$ 3.1

Table 1.2.31

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-68

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
≥ 8	$\leq 24{,}500$	≥ 2.1
≥ 9	$\leq 29{,}500$	≥ 2.4
≥ 11	$\leq 34{,}500$	≥ 2.6
≥ 14	$\leq 39{,}500$	≥ 2.9
≥ 19	$\leq 44{,}500$	≥ 3.0

Table 1.2.32

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-32; PWR FUEL WITH ZR CLADDING AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS ~~(Note 1)~~

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
≥ 12	≤ 24,500	≥ 2.3
≥ 14	≤ 29,500	≥ 2.6
≥ 16	≤ 34,500	≥ 2.9
≥ 19	≤ 39,500	≥ 3.2
≥ 20	≤ 42,500	≥ 3.4

~~NOTES:~~

~~1. MPC-32 is not authorized for transportation at this time.~~

Table 1.2.33

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS FOR TRANSPORTATION IN MPC-32; PWR FUEL WITH ZR CLADDING AND WITH ZIRCALOY IN-CORE GRID SPACERS ~~(Note 1)~~

ASSEMBLY POST-IRRADIATION COOLING TIME (years)	ASSEMBLY BURNUP (MWD/MTU)	ASSEMBLY ENRICHMENT (wt. % ^{235}U)
≥ 8	≤ 24,500	≥ 2.3
≥ 9	≤ 29,500	≥ 2.6
≥ 12	≤ 34,500	≥ 2.9
≥ 14	≤ 39,500	≥ 3.2
≥ 19	≤ 44,500	≥ 3.4

~~NOTES:~~

~~1. MPC-32 is not authorized for transportation at this time.~~

Table 1.2.34

FUEL ASSEMBLY *MAXIMUM ENRICHMENT AND* MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32 ~~(Note 1)~~

FUEL ASSEMBLY ARRAY/CLASS	*Con-figu-ration (Note 2)*	*Maximum Enrichment (wt% ^{235}U)*	*MINIMUM BURNUP (B) AS A FUNCTION OF INITIAL ENRICHMENT (E) (Note 1) (GWD/MTU)*
15x15D, E, F, H	*A*	*4.79*	*B = +(1.1483) * E^3 -(13.4246) * E^2 +(63.2842) * E -71.4084*
	B	*4.54*	*B = +(1.535) * E^3 -(16.895) * E^2 +(73.48) * E -79.05*
	C	*4.64*	*B = +(1.23) * E^3 -(14.015) * E^2 +(64.365) * E -69.9*
	D	*4.59*	*B = +(1.34) * E^3 -(15.13) * E^2 +(68.24) * E -74.07*
17x17A, B, C	*A*	*4.70*	*B = +(0.74) * E^3 -(8.749) * E^2 +(47.7133) * E -57.8113*
	B	*4.31*	*B = +(1.1767) * E^3 -(12.825) * E^2 +(60.7983) * E -67.83*
	C	*4.45*	*B = +(1.3633) * E^3 -(14.815) * E^2 +(66.5517) * E -73.07*
	D	*4.38*	*B = +(1.32) * E^3 -(14.5) * E^2 +(66.39) * E -73.56*

Notes:

~~1. MPC-32 is not authorized for transportation at this time.~~

~~2~~*1*. E = Initial enrichment ~~from the fuel vendor's data sheet~~, i.e., for 4.05wt. %, E = 4.05.
2. See Table 1.2.37

Table 1.2.35

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	**Assembly Burnup (MWD/MTU)**	**Assembly Minimum Enrichment (wt. % ^{235}U)**
$\geq$ 16	$\leq$ 42,000	$\geq$ 3.09
$\geq$ 16	$\leq$ 37,500	$\geq$ 2.6
$\geq$ 16	$\leq$ 30,000	$\geq$ 2.1

Notes:

1. Each fuel assembly must only meet one set of limits (i.e., one row).

Table 1.2.36

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCE COOLING AND BURNUP LIMITS

Type Of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (years)
BPRAs	$\leq 15,998$	≥ 24
TPDs	$\leq 118,674$	≥ 11
RCCAs	$\leq 125,515$	≥ 9
Cf neutron source	$\leq 15,998$	≥ 24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	$\leq 45,361$	≥ 19
Sb-Be neutron source with 4 source rods and 20 thimble plug rods	$\leq 88,547$	≥ 9

Table 1.2.37

~~SOLUBLE BORON REQUIREMENTS FOR MPC-32~~
~~WET LOADING AND UNLOADING OPERATIONS~~LOADING CONFIGURATIONS FOR THE MPC-32

Configuration~~FUEL ASSEMBLY INITIAL ENRICHMENT (wt. % ^{235}U)~~	*Assembly Specifications~~REQUIRED SOLUBLE BORON IN MPC WATER (ppmb)~~*
A~~All fuel assemblies ≤ 4.1~~	o *Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or* o *Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.~~≥ 1900~~*
B~~One or more fuel assemblies > 4.1 and ≤ 5.0~~	o *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank.* o *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.~~≥ 2600~~*
C	o *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWd/mtU of the assembly.* o *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.*
D	o *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWd/mtU of the assembly.* o *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.*

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[1.0.5] NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel",, U.S. Nuclear Regulatory Commission, March 2000.

[1.0.6] HI-STAR 100 Final Safety Analysis Report, Holtec Report No. HI-2012610, Revision 1, Docket No. 72-1008.

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[1.1.2] U.S. Department of Energy, "MPC Transportation Cask Subsystem Design Procurement Specification", Document No. DBF 000000-01717-6300-00001, Rev. 5, January 11, 1996.

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[1.2.7] U.S. DOE SRC/CNEAF/95-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1993, Feb. 1995.

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[1.2.11] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.

[1.2.12] Trojan ISFSI Safety Analysis Report, Revision 3, USNRC Docket 72-0017.

[1.2.13] NRC Interim Staff Guidance Document No. 8, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks", Revision 2.

[1.2.14] NRC Interim Staff Guidance Document No. 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel", Revision 3~~2~~.

[1.2.15] DOE/RW-0184, Volume 3, "Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," Appendix 2.A, "Physical Descriptions of LWR Fuel Assemblies," U.S. Department of Energy, Office of Civilian Radioactive Waste Management, December 1987.

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CHAPTER 3: THERMAL EVALUATION

3.0 INTRODUCTION

In this chapter, compliance of the HI-STAR System thermal performance to 10CFR71 requirements is established for normal transport and hypothetical accident conditions of transport. The analysis considers passive rejection of decay heat from the spent nuclear fuel (SNF) to an environment under the most severe 10CFR71 mandated design basis ambient conditions.

10CFR71 defines the requirements and acceptance criteria that must be fulfilled by the cask thermal design. The requirements and acceptance criteria applicable to the thermal analysis presented in this chapter are summarized here as follows:

1. The applicant must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. [71.33].

   The description must include, with respect to the packaging, specific materials of construction, weights, dimensions, and fabrication methods of materials specifically used as nonfissile neutron absorbers or moderators [71.33(a)(5)(ii)]; and structural and mechanical means for the transfer and dissipation of heat [71.33(a)(5)(v)].

   The description must include, with respect to the contents of the package, chemical and physical form [71.33(b)(3)]; maximum normal operating pressure [71.33(b)(5)]; maximum amount of decay heat [71.33(b)(7)]; and identification and volumes of any coolants [71.33(b)(8)].

2. A package must be designed, constructed, and prepared for shipment so that under normal conditions of transport there would be no substantial reduction in the effectiveness of the packaging [71.43(f) and 71.51(a)(1)].

3. A package must be designed, constructed, and prepared for shipment so that in still air at 100°F and in the shade, no accessible surface of the package would have a temperature exceeding 185°F in an exclusive use shipment [71.43(g)].

4. Compliance with the permitted activity release limits for a Type B package may not depend on filters or on a mechanical cooling system [71.51(c)].

5. With respect to the initial conditions for the events of normal conditions of transport and hypothetical accident conditions, the demonstration of compliance with the requirements of 10CFR71 must be based on the ambient temperature preceding and following the event remaining constant at that value between -20°F and 100°F which is most unfavorable for the feature under consideration. The initial internal pressure within the containment

system must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the event is more unfavorable [71.71(b) and 71.73(b)].

6. For normal conditions of transport, a heat event consisting of an ambient temperature of 100°F in still air and prescribed insolation must be evaluated [71.71(c)(1)].

7. For normal conditions of transport, a cold event consisting of an ambient temperature of -40°F in still air and shade must be evaluated [71.71(c)(2)].

8. Evaluation for hypothetical accident conditions is to be based on sequential application of the specified events, in the prescribed order, to determine their cumulative effect on a package [71.73(a)].

9. For hypothetical accident conditions, a thermal event consisting of a fully engulfing hydrocarbon fuel/air fire with an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 1475°F for a period of 30 minutes [71.73(c)(4)].

As demonstrated in this chapter, the HI-STAR System design and thermal analyses comply with <u>all</u> nine requirements and acceptance criteria listed above. Subsection 3.2 lists the material properties data required to perform the thermal analyses and Subsection 3.3 provides the applicable temperature limits criteria required to demonstrate the adequacy of the HI-STAR System design under all conditions. All thermal analyses to evaluate the normal conditions of transport performance of a HI-STAR System are described in Subsection 3.4. All thermal analyses for hypothetical accident conditions are described in Subsection 3.5. A summary discussion of regulatory compliance is included in Subsection 3.6.

This revision to the HI-STAR transport Safety Analysis Report incorporates certain conforming changes to the multi purpose canisters (MPCs) that are engineered to be transported in the HI-STAR overpack and adoption of ISG-11, Rev. 3 requirements. The principal changes are:

- *The Aluminum Heat Conduction Elements (AHCE) in the MPC, required under CoCs 9261-1 and 9261-2, are rendered optional hardware.*
- *Include a higher capacity PWR basket configuration (MPC-32).*
- *Include an enhanced 24-cell PWR basket layout (MPC-24E), an enlarged cell opening for the MPC-24 and a shortened-height MPC-24E for Trojan fuel.*
- *Raise the nominal helium fill pressure to 42.8 psig.*

- *Relax certain elements of excessive conservatism in the mathematical models to retain a moderate level of conservatism.*
- *The thermal evaluation is revised to comply with the ISG-11, Rev. 3 temperature limits [3.1.5].*
- *Define a "load-and-go" operation wherein only the preferred method of MPC demoisturization - Forced Helium Dehydration (FHD) – is permitted.*

Aside from the above-mentioned changes, this revision of this chapter is essentially identical to its predecessor.

3.3 TECHNICAL SPECIFICATIONS FOR COMPONENTS

HI-STAR System materials and components which are required to be maintained within their safe operating temperature ranges to ensure their intended function are summarized in Table 3.3.1. Long-term stability and continued neutron shielding ability of the Holtite-A neutron shield material under normal transport conditions are ensured when material exposure temperatures are maintained below the maximum allowable limit. The overpack metallic seals will continue to ensure leak tightness of the closure plate, and drain and vent ports if the manufacturer's recommended design temperature limits are not exceeded. Integrity of SNF during transport requires demonstration of HI-STAR System thermal performance to maintain fuel cladding temperatures below design basis limits. Boral used in MPC baskets for criticality control (a composite material composed of B_4C and aluminum) is stable up to 1000°F for short-term and 850°F for long term dry storage†. However, for conservatism, a lower maximum temperature limit is imposed.

Compliance to 10CFR71 requires evaluation of hypothetical accident conditions. The inherent mechanical stability characteristics of the HI-STAR System materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of the HI-STAR System's thermal performance under hypothetical accident conditions, material temperature limits for short-duration events are also provided in Table 3.3.1. *In this Table, the cladding temperature limits of ISG-11, Rev. 3 [3.1.5] are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation. Subsections 3.3.1 through 3.3.3 and their associated tables and figures are no longer needed and are deleted.*

† AAR Structures Boral thermophysical test data.

Table 3.3.1

HI-STAR SYSTEM MATERIAL TEMPERATURE LIMITS

Material	Normal Condition Temperature Limits	Accident Condition Temperature Limits
CSF Cladding	*752°F*	1058°F
Boral†	800°F	950°F
Overpack Closure Plate Mechanical Seals	See Table 4.1.1	See Table 4.1.1
Overpack Vent and Drain Port Plug Seals	See Table 4.1.1	See Table 4.1.1
Aluminum Alloy 5052	176°F††	1105°F†††
Holtite-A	300°F††††	N/A†††††
Aluminum Heat Conduction Elements (Alloy 1100)	725°F	950°F

† Based on AAR Structures Boral thermophysical test data.

†† AL-STAR impact limiter aluminum honeycomb test data.

††† Melting range of alloy is 1105°F-1200°F [3.3.1].

†††† Neutron shield manufacturer's test data (Appendix 1.B).

††††† For shielding analysis (Chapter 5), Holtite-A is conservatively assumed to be lost during the fire accident.

Tables 3.3.2 through 3.3.8

[INTENTIONALLY DELETED]

3.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF TRANSPORT

3.4.1 Thermal Model

The HI-STAR MPC basket designs consist of four distinct geometries engineered to hold 24 and 32 PWR (MPC-24, MPC-24E and MPC-32) or 68 BWR (MPC-68) fuel assemblies. The fuel basket forms a honeycomb matrix of square-shaped fuel compartments to retain the fuel assemblies during transport (refer to Figures 1.2.3 and 1.2.5 for an illustration of PWR and BWR baskets). The basket is formed by an interlocking honeycomb structure of steel plates and full-length edge welding of the cell corners to form an integral basket configuration. Individual cell walls (except outer periphery MPC-68 and MPC-32 cell walls) are provided with Boral neutron absorber panels, which consists of a Boral plate sandwiched between the cell wall and a stainless steel sheathing plate, for the full length of the active fuel region.

The design basis decay heat generation per PWR or BWR assembly for normal transport for each MPC type is specified in Table 1.2.13. The decay heat is considered to be nonuniformly distributed over the active fuel length based on the design basis axial burnup distribution specified in Chapter 1 (see Table 1.2.15 and Figures 1.2.13 and 1.2.14).

Transport of heat from the MPC basket interior to the basket periphery is accomplished by conduction through the MPC basket metal grid structure and the narrow helium gaps between the fuel assemblies and fuel cell walls. Heat dissipation in the MPC basket periphery-to-MPC shell gap is by a combination of helium conduction, natural convection (by means of the "Rayleigh" effect) and radiation across the gap. Between the MPC shell and the overpack inner shell is a small clearance which is evacuated and backfilled with helium. Helium, besides being inert, is a better conductor of heat than air. Thus, heat conduction through the helium gap between the MPC and the overpack will minimize temperature differentials across this region.

The overpack, under normal transport conditions, passively rejects heat to the environment. Cooling of the exterior system surfaces is by natural convection and radiation. During transport, the HI-STAR System is placed in a horizontal position with stainless steel encased aluminum honeycomb impact limiters installed at both ends of the overpack. To conservatively maximize the calculated internal temperatures, the thermal conductivity of the impact limiters is set essentially equal to zero. Under normal transport conditions, the MPC shell rests on the overpack internal cavity surface forming an eccentric gap. Direct contact between the MPC and overpack surfaces is expected to minimize heat transfer resistance in this region of intimate contact. Significantly improved conductive heat transport due to reduction in the helium gap near the contact region is accounted for in the thermal analysis of the HI-STAR System. The HI-STAR System is conservatively analyzed assuming a minimum 0.02-inch gap at the line of metal-to-metal contact. Analytical modeling details of the various thermal transport mechanisms are provided in the following.

3.4.1.1 Analytical Model - General Remarks

Transport of heat from the heat generation region (fuel assemblies) to the outside environment is analyzed broadly in terms of three interdependent thermal models.

i. The first model considers transport of heat from the fuel assembly to the basket cell walls. This model recognizes the combined effects of conduction (through helium) and radiation, and is essentially a finite element technology-based update of the classical Wooton & Epstein [3.4.1] formulation (which considers radiative heat exchange between fuel rod surfaces).

ii. The second model considers heat transport within an MPC cross section by conduction and radiation. The effective cross sectional thermal conductivity of the basket region obtained from the combined fuel assembly/basket heat conduction radiation model is applied to an axisymmetric thermal model of the HI-STAR System on the FLUENT [3.1.2] code.

iii. The third model deals with the transmission of heat from the MPC exterior surface to the external environment (heat sink). From the MPC shell to the cask exterior surface, heat is conducted through an array of concentric shells representing the MPC-to-overpack helium gap, the overpack inner shell, the intermediate shells, the Holtite-A neutron shielding and finally the overpack outer shell. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the exposed cask surfaces. Insolation on exposed cask surfaces is based on 12-hour levels prescribed in 10CFR71, averaged over a 24-hour period.

The following subsections contain a systematic description of the mathematical models devised to articulate the temperature field in the HI-STAR System. Table 3.4.2 shows the relationship between the mathematical models and the corresponding regions (i.e., fuel, MPC, overpack, etc.) of the HI-STAR System. The description begins with the method to characterize the heat transfer behavior of the prismatic (square) opening referred to as the "fuel space" containing a heat emitting fuel assembly. The methodology utilizes a finite-volume procedure to replace the heterogeneous SNF/fuel space region with an equivalent solid body having a well-defined temperature-dependent conductivity. In the following subsection, the method to replace the composite walls of the fuel basket cells with equivalent "solid" walls is presented. Having created the mathematical equivalents for the SNF/fuel spaces and the fuel basket walls, the method to represent the MPC cylinder containing the fuel basket by an equivalent cylinder whose thermal conductivity is a function of the spatial location and coincident temperature is presented.

Following the approach of presenting descriptions starting from the inside and moving to the outer region of a cask, the next subsections present the mathematical model to simulate the overpack. Subsection 3.4.1.1.12 concludes the presentation with a description of how the different models for the specific regions within the HI-STAR System are assembled into the final finite element model.

3.4.1.1.1 Overview of the Thermal Model

Thermal analysis of the HI-STAR System is performed by assuming that the system is subject to its maximum heat duty with each storage location occupied and with the heat generation rate in each stored fuel assembly equal to the design basis maximum value. While the assumption of equal heat generation imputes a certain symmetry to the cask thermal problem, the thermal model must incorporate three attributes of the physical problem to perform a rigorous analysis:

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

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

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

It is clearly impractical to explicitly model every fuel rod in every stored fuel assembly explicitly. Instead, the cross section bounded by the inside of the storage cell, which surrounds the assemblage of fuel rods and the interstitial helium gas, is replaced with an "equivalent" square (solid) section characterized by an effective thermal conductivity. Figure 3.4.1 pictorially illustrates the homogenization concept. Further details on this process for determining the effective conductivity is presented in Subsection 3.4.1.1.2. It suffices to state here that the effective conductivity of the cell space will be a function of temperature, because radiation heat transfer (a major component of the heat transport mechanism between the fuel rods to the basket metal square) is a strong function of the absolute temperatures of the participating bodies. Therefore, in effect, every storage cell location will have a different value of effective conductivity in the homogenized model. The process of determining the temperature-dependent effective conductivity is carried out using a finite volume procedure.

In the next step of homogenization, a planar section of MPC is considered. With each storage cell inside space replaced with an equivalent solid square, the MPC cross section consists of a metallic gridwork (basket cell walls with each cell space containing a solid fuel square with an effective thermal conductivity) circumscribed by a circular ring (MPC shell). There are four principal materials in this section that are included in all MPCs, namely the homogenized fuel cell squares, the Alloy X MPC structural materials in the MPC (including Boral sheathing material), Boral and helium gas. Aluminum heat conduction elements (AHCEs), included optionally in the MPC design, are appropriately ignored in the heat dissipation calculations. Each of the four constituent materials in this section has a different conductivity. As discussed earlier, the conductivity of the homogenized fuel cell is a strong function of temperature.

In order to replace this thermally heterogeneous MPC section with an equivalent conduction-only lamina, resort to the finite-element procedure is necessary. Because the rate of transport of heat within the MPC is influenced by radiation, which is a temperature-dependent effect, the equivalent conductivity of the MPC lamina must be computed as a function of temperature. Finally, it is recognized that the MPC section consists of two discrete regions, namely, the basket region and the periphery region. The periphery region is the space between the peripheral storage cells and the MPC enclosure shell. This space is essentially full of helium gas surrounded by Alloy X plates and optionally aluminum heat conduction elements. Accordingly, as illustrated in Figure 3.4.2 for MPC-68, the MPC cross section is replaced with two homogenized regions with temperature-dependent conductivities. In particular, the effective conductivity of the fuel cells is subsumed into the equivalent conductivity of the basket cross section using a finite element procedure. The ANSYS finite-element code is the vehicle for all modeling efforts described in the foregoing.

In summary, appropriate finite element models are used to replace the MPC cross section with an equivalent two-region homogeneous conduction lamina whose local conductivity is a known function of coincident absolute temperature. Thus, the MPC cylinder containing discrete fuel assemblies, helium, Boral, Alloy X and optionally AHCEs* is replaced with a right circular cylinder whose material conductivity will vary with radial and axial position as a function of the coincident temperature.

The MPC-to-overpack gap is simply an annular space that is readily modeled with an equivalent conductivity that reflects the conduction and radiation modes of heat transfer. The overpack is a radially symmetric structure except for the neutron absorber region which is built from radial connectors and Holtite. Using the classical equivalence procedure as described in Section 3.4.1.1.9, this region is replaced with an equivalent radially symmetric annular cylinder.

The thermal analysis procedure described above makes frequent use of equivalent thermal properties to ease the geometric modeling of the cask components. These equivalent properties are rigorously calculated values based on detailed evaluations of actual cask system geometries. All these calculations are performed conservatively to ensure a bounding representation of the cask system. This process, commonly referred to as submodeling, yields accurate (not approximate) results. Given the detailed nature of the submodeling process, experimental validation of the individual submodels is not necessary.

In this manner, a HI-STAR System overpack containing a loaded MPC is replaced with a right circular cylinder with spatially varying temperature-dependent conductivity. Heat is generated within the basket space in this cylinder in the manner of the prescribed axial distribution. In addition, heat is deposited from insolation on its external surface. Natural convection and thermal radiation to ambient air dissipate heat. Details of the elements of mathematical modeling are provided in the following sections.

* In the thermal modeling, AHCEs are appropriately ignored.

3.4.1.1.2 Fuel Region Effective Thermal Conductivity Calculation

Thermal properties of a large number of PWR and BWR fuel assembly configurations manufactured by the major fuel suppliers (i.e., Westinghouse, CE, B&W, and GE) have been evaluated for inclusion in the HI-STAR System thermal analysis. Bounding PWR and BWR fuel assembly configurations are determined using the simplified procedure described below. This is followed by the determination of temperature-dependent properties of the bounding PWR and BWR fuel assembly configurations to be used for cask thermal analysis using a finite-volume (FLUENT) approach.

To determine which of the numerous PWR assembly types listed in Table 3.4.4 should be used in the thermal model for the PWR fuel baskets, we must establish which assembly has the maximum thermal resistance. The same determination must be made for the MPC-68, out of the menu of SNF types listed in Table 3.4.5. For this purpose, we utilize a simplified procedure that we describe below.

Each fuel assembly consists of a large array of fuel rods typically arranged on a square layout. Every fuel rod in this array is generating heat due to radioactive decay in the enclosed fuel pellets. There is a finite temperature difference required to transport heat from the innermost fuel rods to the storage cell walls. Heat transport within the fuel assembly is based on principles of conduction heat transfer combined with the highly conservative analytical model proposed by Wooton and Epstein [3.4.1]. The Wooton-Epstein model considers radiative heat exchange between individual fuel rod surfaces as a means to bound the hottest fuel rod cladding temperature.

Transport of heat energy within any cross section of a fuel assembly is due to a combination of radiative energy exchange and conduction through the helium gas that fills the interstices between the fuel rods in the array. With the assumption of uniform heat generation within any given horizontal cross section of a fuel assembly, the combined radiation and conduction heat transport effects result in the following heat flow equation:

$$Q = \sigma\, C_o\, F_\varepsilon\, A\, [T_C^4 - T_B^4] + 13.5740\, L\, K_{cs}\, [T_C - T_B]$$

where,

$$F_\varepsilon = \textit{Emissivity Factor} = \frac{1}{\left(\frac{1}{\varepsilon_C} + \frac{1}{\varepsilon_B} - 1\right)}$$

ε_C, ε_B = emissivities of fuel cladding, fuel basket (see Table 3.2.4)

$C_o = Assembly\ Geometry\ Factor$

$$= \frac{4N}{(N+1)^2} (when\ N\ is\ odd)$$

$$= \frac{4}{N+2} (when\ N\ is\ even)$$

N	=	Number of rows or columns of rods arranged in a square array
A	=	fuel assembly "box" heat transfer area
	=	4 × width × length (ft^2)
L	=	fuel assembly length (ft)
K_{cs}	=	fuel assembly constituent materials volume fraction weighted mixture conductivity (Btu/ft-hr-°F)
T_C	=	hottest fuel cladding temperature (°R)
T_B	=	box temperature (°R)
Q	=	net radial heat transport from the assembly interior (Btu/hr)
σ	=	Stefan-Boltzman Constant (0.1714×10^{-8} Btu/ft^2-hr-°R^4)

In the above heat flow equation, the first term is the Wooten-Epstein radiative heat flow contribution while the second term is the conduction heat transport contribution based on the classical solution to the temperature distribution problem inside a square shaped block with uniform heat generation [3.4.3]. The 13.574 factor in the conduction term of the equation is the shape factor for two-dimensional heat transfer in a square section. Planar fuel assembly heat transport by conduction occurs through a series of resistances formed by the interstitial helium fill gas, fuel cladding and enclosed fuel. An effective planar mixture conductivity is determined by a volume fraction weighted sum of the individual constituent materials resistances. For BWR assemblies, this formulation is applied to the region inside the fuel channel. A second conduction and radiation model is applied between the channel and the fuel basket gap. These two models are combined, in series, to yield a total effective conductivity.

The effective thermal conductivities of several representative intact PWR and BWR assemblies are presented in Tables 3.4.4 and 3.4.5. At higher temperatures (greater than 450°F), the zircaloy clad fuel assemblies with the lowest effective thermal conductivities are the Westinghouse 17×17 OFA (PWR) and the General Electric GE-11 9×9 (BWR). A discussion of fuel assembly conductivities for some of the newer 10×10 array *and plant specific* BWR fuel designs is presented near the end of this subsection. Based on this *simplified* analysis, the Westinghouse 17×17 OFA PWR and GE-11 9×9

BWR fuel assemblies are determined to be the bounding configurations for analysis at design basis maximum heat loads. As discussed in Section 3.3.2, stainless clad fuel assemblies with significantly lower decay heat emission characteristics are not deemed to be bounding.

Several of the assemblies listed in Tables 3.4.5 were excluded from consideration when determining the bounding assembly because of their extremely low decay heat loads. The excluded assemblies, which were each used at a single reactor only, are physically small and have extremely low burnups and long cooling times. These factors combine to result in decay heat loads that are much lower than the design basis maximum. The excluded assemblies are:

Dresden Unit 1 8×8
Dresden Unit 1 6×6
Allis-Chalmers 10×10 Stainless
Exxon Nuclear 10×10 Stainless
Humboldt Bay 7x7
Quad$^+$ 8x8

The Allis-Chalmers and Exxon assemblies are used only in the LaCrosse reactor of the Dairyland Power Cooperative. The design basis assembly decay heat loads for Dresden Unit 1 and LaCrosse SNF (Tables 1.2.14 and 1.2.19) are approximately 58% lower and 69% lower, respectively, than the MPC-68 design basis assembly maximum heat load (Table 1.2.13). Examining Table 3.4.5, the effective thermal conductivity of damaged Dresden Unit 1 fuel assemblies inside DFCs (the lowest of any Dresden Unit 1 assembly) and LaCrosse fuel assemblies are approximately 40% lower and 30% lower, respectively, than that of the bounding (GE-11 9×9) fuel assembly. Consequently, the fuel cladding temperatures in the HI-STAR System with Dresden Unit 1 and LaCrosse fuel assemblies (intact or damaged) will be bounded by design basis fuel cladding temperatures.

To accommodate Trojan Nuclear Plant (TNP) SNF in a HI-STAR System's MPC-24E canister*, the discharged fuel characteristics at this permanently shutdown site are evaluated herein. To permit TNP fuel in the HI-STAR System, it is necessary to confirm that certain key fuel parameters, viz. burnup (B) and cask decay heat (D) are bounded by the thermal design limits (42,500 MWD/MTU and 20 kW for PWR MPCs). The TNP SNF is a member of the 17x17 class of fuel types. The bulk of the fuel inventory is from Westinghouse and balance from B&W. The B&W SNF configuration and cladding dimensions are same as that of the Westinghouse 17x17 SNF. The fuel is more than nine years old and the burnups are in the range of 5073 MWD/MTU to 41889 MWD/MTU. The TNP SNF burnups are bounded by the design maximum for PWR class of fuel (i.e. $B < 42500$ MWD/MTU). Because the fuel decay heat is exponentially attenuating with time, it is conservative to evaluate decay heat on a date that precedes fuel loading. For this purpose, a reference date (RD) of 11/9/2001 is employed herein. The decay heat from the most emissive Trojan fuel is bounded by 725 W on RD. Postulating every cell location in an MPC-24E is occupied by this most heat emissive

* The height of MPC-24E for Trojan SNF is shorter than the height of generic HI-STAR MPCs.

fuel assembly, a conservatively bounding D = 17.4 kW* is computed. The Trojan MPC-24E heat loads are below the HI-STAR System design heat load (i.e. D < 20 kW) by a significant margin.

A limited number of Trojan assemblies have poison inserts (RCCAs and BPRAs) and other non-fuel hardware (Thimble Plugs). The inclusion of PWR non-fuel hardware influences the MPC thermal response in two ways: (i) The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures and (ii) Volume displaced by the mass of non-fuel hardware lowers the available cavity free volume for accommodating gas released in hypothetical rod rupture scenarios. For a conservatively bounding evaluation, the thermal modeling ignores the presence of non-fuel hardware and the MPC cavity volume is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included.

Having established the governing (most resistive) PWR and BWR SNF types, a finite-volume code is used to determine the effective conductivities in a conservative manner. Detailed conduction-radiation finite-volume models of the bounding PWR and BWR fuel assemblies are developed in the FLUENT code as shown in Figures 3.4.7 and 3.4.8, respectively. The PWR model was originally developed on the ANSYS code which enables individual rod-to-rod and rod-to-basket wall view factor calculations to be performed using that code's AUX12 processor. Limitations of radiation modeling techniques implemented in ANSYS make it difficult to take advantage of the symmetry of the fuel assembly geometry. Unacceptably long CPU time and large workspace requirements necessary for performing gray body radiation calculations for a complete fuel assembly geometry on ANSYS prompted the development of an alternate simplified model on the FLUENT code. The FLUENT model was benchmarked with the ANSYS model results for a Westinghouse 17×17 OFA fuel assembly geometry for the case of black body radiation (emissivities = 1). The FLUENT model was found to yield conservative results in comparison to the ANSYS model for the "black" surface case. The FLUENT model benchmarked in this manner is used to solve the gray body radiation problem to provide the necessary results for determining the effective thermal conductivity of the governing PWR fuel assembly. The same modeling approach using FLUENT is then applied to the governing BWR fuel assembly and the effective conductivity of GE-11 9×9 fuel is determined.

An equivalent homogeneous material that fills the basket opening replaces the combined fuel rods-helium matrix by the following two-step procedure. In the first step, the FLUENT-based fuel assembly model is solved by applying equal heat generation per unit length to the individual fuel rods and a uniform boundary temperature along the basket cell opening inside periphery. The temperature difference between the peak cladding and boundary temperatures is used to determine an effective conductivity as described in the next step. For this purpose, we consider a two-dimensional cross section of a square shaped block of size equal to 2L and a uniform volumetric heat source (q_g) cooled at the periphery with a uniform boundary temperature. Under the assumption of constant material thermal conductivity (K), the temperature difference (ΔT) from the center of the cross section to the periphery is analytically given by [3.4.3]:

* Projected MPC heat loads are much lower (in the range of 6 kw to 14.5 kW in circa 2003).

$$\Delta T = 0.29468 \frac{q_g L^2}{K}$$

This analytical formula is applied to determine the effective material conductivity from a known quantity of heat generation applied in the FLUENT model (smeared as a uniform heat source, q_g), basket opening size and ΔT calculated in the first step.

As discussed earlier, the effective fuel space conductivity is a function of the temperature coordinate. The above two step analysis is carried out for a number of reference temperatures. In this manner, the effective conductivity as a function of temperature is established.

In Table 3.4.25, 10×10 array type BWR fuel assembly effective thermal conductivity results from a simplified analysis are presented to determine the most resistive fuel assembly in this class. Using the simplified analysis procedure discussed earlier, the Atrium-10 fuel type is determined to be the most resistive in this class of fuel assemblies. A detailed finite-element model of this assembly type was developed to rigorously quantify the heat dissipation characteristics. The results of this study are presented in Table 3.4.26 and compared to the bounding BWR fuel assembly effective thermal conductivity depicted in Figure 3.4.13. The results of this study demonstrate that the bounding BWR fuel assembly effective thermal conductivity is conservative with respect to the 10×10 class of BWR assemblies. Table 3.4.34 summarizes plant specific fuel types' effective conductivities. From these analytical results, the SPC-5 is determined to be the most resistive fuel assembly in this group of fuel types. A rigorous finite element model of SPC-5 fuel assembly was developed to confirm that its in-plane heat dissipation characteristics are bounded from below by the design basis BWR fuel conductivities used in the HI-STAR thermal analysis.

Temperature-dependent effective conductivities of PWR and BWR design basis fuel assemblies (most resistive SNF types) are shown in Figure 3.4.13. The finite-volume results are also compared to results reported from independent technical sources. From this comparison, it is readily apparent that FLUENT-based fuel assembly conductivities are conservative. The FLUENT computed values (not the published literature data) are used in the MPC thermal analysis presented in this document.

3.4.1.1.3 Effective Thermal Conductivity of Sheathing/Boral/Cell Wall Sandwich

Each MPC basket cell wall (except outer periphery MPC-68 & mpc-32 cell walls) is manufactured with a Boral neutron absorbing plate for criticality control. Each Boral plate is sandwiched in a sheathing-to-basket wall pocket. A schematic of the "Box Wall-Boral-Sheathing" sandwich geometry of an MPC basket is illustrated in Figure 3.4.5. During fabrication, a uniformly applied normal pressure on each sheathing-Boral-cell wall sandwich prior to stitch welding of the sheathing periphery to the box wall ensures adequate surface-to-surface contact for elimination of any macroscopic gaps. The mean coefficient of linear expansion of Boral is higher than the basket materials thermal expansion coefficients. Consequently, basket heat-up from the contained SNF will further ensure a tight fit of the Boral plate in the sheathing-to-cell wall pocket. The presence of small microscopic gaps due to less than perfect surface finish characteristics requires consideration of an interfacial contact resistance between the Boral and the box and sheathing surfaces. A conservative contact resistance resulting from a 2 mils Boral-to-pocket gap is applied to the analysis. Note that this gap would actually be filled with helium. In other words, no credit is taken for the interfacial pressure between Boral and stainless plate/sheet stock produced by the fixturing and welding process.

Heat conduction properties of a composite "Box Wall-Boral-Sheathing" sandwich in the two principal basket cross sectional directions as illustrated in Figure 3.4.5 (i.e., lateral "out-of-plane" and longitudinal "in-plane") are unequal. In the lateral direction, heat is transported across layers of sheathing, helium-gap, Boral (B_4C and cladding layers) helium-gap, and cell wall resistances that are in series (except for the small helium filled end regions shown in Figure 3.4.6). Heat conduction in the longitudinal direction, in contrast, is through an array of essentially parallel resistances comprised of these same layers. For the ANSYS based MPC basket thermal model, corresponding non-isotropic effective thermal conductivities in the two orthogonal directions are determined and applied in the analysis.

The non-isotropic conductivities are determined by constructing ANSYS models of the composite "Box Wall-Boral-Sheathing" sandwich for the "in-plane" and "out-of-plane" directions. For determining the effective conductivity (K_{eff}), a heat flux is applied to the to one end of the sandwich and an ANSYS numerical solution to the sandwich temperature differential obtained. From Fourier equation for one-dimensional conduction heat transfer, the following equation for K_{eff} is obtained:

$$K_{eff} = \frac{qL}{\Delta T}$$

where:

q = Sandwich heat flux
L = Sandwich length in the direction of heat transfer
ΔT = Sandwich temperature differential (obtained from ANSYS solution)

In the equation above, L is the width or thickness of the sandwich, respectively, for in-plane or out-of-plane heat transfer directions.

3.4.1.1.4 Modeling of Basket Conductive Heat Transport

Conduction of heat in a fuel basket is a combination of planar and axial contributions. These component contributions are individually calculated for each MPC basket design and combined (as described later in this subsection) to obtain an equivalent isotropic thermal conductivity. The heat rejection capability of each MPC design (i.e., MPC-24, MPC-24E, MPC-32 and MPC-68) is evaluated by developing a thermal model of the combined fuel assemblies and composite basket walls geometry on the ANSYS finite element code. The ANSYS model includes a geometric layout of the basket structure in which the "Box Wall-Boral-Sheathing" sandwich is replaced by a "homogeneous wall" with an equivalent thermal conductivity. Since the thermal conductivity of the Alloy X material is a weakly varying function of temperature, the equivalent "homogeneous wall" must have a temperature-dependent effective conductivity. Similarly, as illustrated in Figure 3.4.6, the conductivities in the in-plane and through-thickness direction of the equivalent "homogeneous wall" are different. Finally, as discussed earlier, the fuel assemblies occupying the basket cell openings are modeled as homogeneous heat generating regions with effective temperature dependent in-plane conductivities. The methodology used to reduce the heterogeneous MPC basket - fuel assemblage to an equivalent homogeneous region with effective thermal properties is discussed in the following.

Consider a cylinder of height L and radius r_o with a uniform volumetric heat source term q_g, with insulated top and bottom faces and its cylindrical boundary maintained at a uniform temperature T_c. The maximum centerline temperature (T_h) to boundary temperature difference is readily obtained from classical one-dimensional conduction relationships (for the case of a conducting region with constant thermal conductivity K_s):

$$(T_h - T_c) = q_g\, r_o^2/(4\, K_s)$$

Noting that the total heat generated in the cylinder (Q_t) is $\pi\, r_o^2\, L\, q_g$, the above temperature rise formula can be reduced to the following simplified form in terms of the total heat generation per unit length (Q_t/L):

$$(T_h - T_c) = (Q_t / L)/ (4\, \pi\, K_s)$$

This simple analytical approach is employed to determine an effective basket cross-sectional conductivity by applying an equivalence between the ANSYS finite element model of the basket and the analytical case. The equivalence principle employed in the HI-STAR System thermal analysis is depicted in Figure 3.4.2. The 2-dimensional ANSYS finite element model of the MPC basket is solved by applying a uniform heat generation per unit length in each basket cell region and a constant basket periphery boundary temperature, T_c'. Noting that the basket region with uniformly distributed heat sources and a constant boundary temperature is equivalent to the analytical case of a cylinder with uniform volumetric heat source discussed earlier, an effective MPC basket conductivity (K_{eff}) is readily derived from the analytical formula and the ANSYS solution leading to the following relationship:

$$K_{eff} = N\, (Q_f'/L) / (4\, \pi\, [T_h' - T_c'])$$

where:

N = number of fuel assemblies

(Q_f'/L) = each fuel assembly heat generation per unit length applied in ANSYS model

T_h' = peak basket cross-section temperature from ANSYS model

Cross sectional views of MPC basket ANSYS models are illustrated in Figures 3.4.10 and 3.4.11 for a PWR and BWR MPC. Notice that many of the basket supports and all shims have been conservatively neglected in the models. This conservative geometry simplification, coupled with the conservative neglect of thermal expansion which would minimize the gaps, yields conservative gap thermal resistances. Temperature dependent equivalent thermal conductivities of the fuel region and composite basket walls, as determined from analysis procedures described earlier, are applied to the ANSYS model. The planar ANSYS conduction model is solved by applying a constant basket periphery temperature with uniform heat generation in the fuel region. Table 3.4.6 summarizes effective thermal conductivity results of each basket design obtained from the ANSYS models. It is recalled that the equivalent thermal conductivity values presented in Table 3.4.6 are lower bound values because, among other elements of conservatism, the effective conductivity of the most resistive SNF type (Tables 3.4.4 and 3.4.5) is used in the MPC finite-element simulations.

The axial conductivity of a fuel basket is determined by calculating a cross-sectional area-weighted sum of the component conductivities (Helium, Alloy-X, Boral and fuel cladding). In accordance with NUREG-1536 guidelines, credit for fuel rod axial heat conduction is conservatively limited to cladding.

Having obtained planar and axial thermal conductivities as described above, an equivalent isotropic conductivity (defined as the Square Root of the Mean Sum of Squares (SRMSS*)) is obtained as shown below:

$$k_{iso} = \sqrt{\frac{k_{rad}^2 + k_{ax}^2}{2}}$$

where:

- k_{iso} = equivalent isotropic thermal conductivity
- k_{rad} = equivalent planar thermal conductivity
- k_{ax} = equivalent axial thermal conductivity

The equivalent isotropic conductivities are employed in the HI-STAR thermal modeling as discussed in Subsection 3.4.2.

3.4.1.1.5 Heat Transfer in MPC Basket Peripheral Regions

Each of the MPC designs for storing PWR or BWR fuel are provided with relatively large helium filled regions formed between the relatively cooler MPC shell and hot basket peripheral panels. For a horizontally oriented cask under normal transport conditions, heat transfer in these helium-filled

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* This formulation has been benchmarked for specific application to the MPC basket designs and confirmed to yield conservative results.

regions is similar to heat transfer in closed cavities under three cases listed below:

i. differentially heated short vertical cavity

ii. horizontal channel heated from below

iii. horizontal channel heated from above

In a closed cavity (case i scenario), an exchange of hot and cold fluids occurs near the top and bottom ends of the cavity, resulting in a net transport of heat across the gap.

The case (ii) scenario is similar to the classical Rayleigh-Benard instability of a layer of fluid heated from below [3.4.6]. If the condition for onset of fluid motion is satisfied, then a multi-cellular natural convection pattern is formed. The flow pattern results in upward motion of heated fluid and downward motion of relatively cooler fluid from the top plate, resulting in a net transport of heat across the heated fluid channel.

The case (iii) is a special form of case (ii) with an inverted (stably stratified) temperature profile. No fluid motion is possible in this circumstance and heat transfer is thus limited to fluid (helium) conduction only.

The three possible cases of closed cavity natural convection are illustrated in Figure 3.4.3 for an MPC-68 basket geometry. Peripheral spaces labeled B and B' illustrate the case (i) scenario, the space labeled D illustrates the case (ii) scenario, and the space labeled D' illustrates the case (iii) scenario. The basket is oriented to conservatively maximize the number of peripheral spaces having *no* fluid motion. A small alteration in the basket orientation will result in a non-zero gravity component in the x-direction which will induce case (i) type fluid motion in the D' space. The rate of natural convection heat transfer is characterized by a Rayleigh number for the cavity defined as follows:

$$Ra_L = \frac{C_p \rho^2 g \beta \Delta T L^3}{\mu K}$$

where:

C_p	=	fluid heat capacity
ρ	=	average fluid density
g	=	acceleration due to gravity
β	=	coefficient of thermal expansion (equal to reciprocal of absolute temperature for gases)

ΔT	=	temperature difference between hot and cold surfaces
L	=	spacing between hot and cold surfaces
μ	=	fluid viscosity
K	=	fluid conductivity

Hewitt et al. [3.4.5] report Nusselt number correlations for the closed cavity natural convection cases discussed earlier. A Nusselt number equal to unity implies heat transfer by fluid conduction only. A higher than unity Nusselt number is due to the so-called "Rayleigh" effect, which monotonically rises with increasing Rayleigh number. Nusselt numbers applicable to helium filled PWR and BWR MPCs in the peripheral voids are provided in Table 3.4.1. For *conservatively maximizing HI-STAR normal transport temperatures* ~~conservatism, the~~ heat dissipation enhancement due to Rayleigh effect is ignored.

3.4.1.1.6 Effective Conductivity of Multi-Layered Intermediate Shell Region

Fabrication of the layered overpack intermediate shells is discussed in Section 1.2 of this SAR. In the thermal analysis, each intermediate shell metal-to-metal interface presents an additional resistance to heat transport. The contact resistance arises from microscopic pockets of air trapped between surface irregularities of the contacting surfaces. Since air is a relatively poor conductor of heat, this results in a reduction in the ability to transport heat across the interface compared to that of the base metal. Interfacial contact conductance depends upon three principal factors, namely: (i) base material conductivity, (ii) interfacial contact pressure, and (iii) surface finish.

Rohsenow and Hartnett [3.2.2] have reported results from experimental studies of contact conductance across air entrapped stainless steel surfaces with a typical 100 μ-inch surface finish. A minimum contact conductance of 350 Btu/ft-hr-°F is determined from extrapolation of results to zero contact pressure.

The thermal conductivity of carbon steel is about three times that of stainless steel. Thus the choice of carbon steel as the base material in a multi-layered construction significantly improves heat transport across interfaces. The fabrication process guarantees interfacial contact. Contact conductance values extrapolated to zero contact pressures are therefore conservative. The surface finish of hot-rolled carbon steel plate stock is generally in the range of 250-1000 μ-inch [3.2.1]. The process of forming hot-rolled flat plate stock to cylindrical shapes to form the intermediate shells by rolling will result in a smoother surface finish. This results from the large surface pressures exerted by the hardened roller faces that flatten out any surface irregularities.

In the HI-STAR thermal analysis, a conservatively bounding interfacial contact conductance value is determined based on the following assumptions:

1. No credit is taken for high base metal conductivity.
2. No credit is taken for interfacial contact pressure.
3. No credit is taken for a smooth surface finish resulting from rolling of hot-rolled plate stock to cylindrical shapes.
4. Contact conductance is based on a uniform 2000 μ-inch (1000 μ-inch for each surface condition) interfacial air gap at all interfaces.
5. No credit for radiation heat exchange across this hypothetical inter-surface air gap.
6. Bounding low thermal conductivity at 200°F.

These assumptions guarantee a conservative assessment of heat dissipation characteristics of the multi-layered intermediate shell region. The resistances of the five carbon steel layers along with the associated interfacial resistances are combined as resistances in series to determine an effective conductivity of this region leading to the following relationship:

$$K_{gs} = r_o \ell n\left[\frac{r_5}{r_o}\right]\left[\sum_{i=1}^{5} \frac{\delta}{K_{air}} \frac{r_o}{r_i} + \frac{r_o \ell n\left[\frac{r_5}{r_o}\right]}{K_{cst}}\right]^{-1}$$

where (in conventional U.S. units):

K_{gs}	=	effective intermediate shell region thermal conductivity
r_o	=	inside radius of inner gamma shield layer
r_i	=	outer radius of i^{th} intermediate shell layer
δ	=	interfacial air gap (2000 μ-inch)
K_{air}	=	air thermal conductivity
K_{cst}	=	carbon steel thermal conductivity

3.4.1.1.7 Heat Rejection from Overpack and Impact Limiter Outside Surfaces

Jakob and Hawkins [3.2.9] recommend the following correlations for natural convection heat transfer to air from heated vertical surfaces (flat impact limiter ends) and from single horizontal cylinders (overpack and impact limiter curved surfaces):

Turbulent range:

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

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

(in conventional U.S. units)

Laminar range:

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

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

(in conventional U.S. units)

where ΔT is the temperature differential between the system exterior surface and ambient air. During normal transport conditions, the surfaces to be cooled are the impact limiter and overpack cylindrical surfaces, and the flat vertical faces of the impact limiters. The corresponding length scales for these surfaces are the impact limiter diameter, overpack diameter, and impact limiter diameter, respectively. Noting that Gr×Pr is expressed as $L^3 \Delta T Z$, where Z (from Table 3.2.7) is at least 2.6×10^5 at a conservatively high upper bound system exterior surface temperature of 340°F, it is apparent that the turbulent condition is always satisfied for ΔT in excess of a few degrees Fahrenheit. Under turbulent conditions, the more conservative heat transfer correlation for horizontal cylinders (i.e., $h = 0.18\ \Delta T^{1/3}$) is utilized for thermal analyses on all exposed system surfaces.

Including both convective and radiative heat loss from the system exterior surfaces, the following relationship for surface heat flux is developed:

$$q_s = 0.18\,(T_s - T_A)^{4/3} + \sigma \times \varepsilon \times [(T_s + 460)^4 - (T_A + 460)^4]$$

where:

T_s, T_A = surface, ambient temperatures (°F)
q_s = surface heat flux (Btu/ft^2-hr)
ε = surface emissivity (see Table 3.2.4)
σ= Stefan-Boltzman Constant (0.1714×10^{-8} Btu/ft^2-hr-°R^4)

3.4.1.1.8 Determination of Solar Heat Input

The intensity of solar radiation incident on an exposed surface depends on a number of time varying parameters. The solar heat flux strongly depends upon the time of the day as well as on latitude and day of the year. Also, the presence of clouds and other atmospheric conditions (dust, haze, etc.) can significantly attenuate solar intensity levels. Rapp [3.4.2] has discussed the influence of such factors in considerable detail.

The HI-STAR System thermal analysis is based upon insolation levels specified in 10CFR71, Subpart F, which are for a 12-hour daytime period. During normal transport conditions, the HI-STAR System is cyclically subjected to solar heating during the 12-hour daytime period followed by cooling during the 12-hour nighttime. However, due to the large mass of metal and the size of the system, the inherent dynamic time lag in the temperature response is substantially larger than the 24-hour heating-cooling time period. Accordingly, the HI-STAR System cask model includes insolation at exposed surfaces averaged over a 24-hour time period. A bounding solar absorption coefficient of 1.0 is applied to cask exterior surfaces. The 10CFR71 mandated 12-hour average incident solar radiation levels are summarized in Table 3.4.7. The combined incident insolation heat flux absorbed by exposed cask surfaces and decay heat load from the MPC is rejected by natural convection and radiation to ambient air.

3.4.1.1.9 Effective Thermal Conductivity of Radial Channels - Holtite Region

In order to minimize heat transfer resistance limitations due to the poor thermal conductivity of the Holtite-A neutron shield material, a large number of thick radial channels formed from high strength and conductivity carbon steel material are embedded in the neutron shield region. These radial channels form highly conductive heat transfer paths for efficient heat removal. Each channel is welded to the outside surface of the outermost intermediate shell and at the overpack enclosure shell, thereby providing a continuous path for heat removal to the ambient environment.

The effective thermal conductivity of the composite neutron shielding and radial channels region is determined by combining the heat transfer resistance of individual components in a parallel network. In determining the heat transfer capability of this region to the outside ambient environment for normal transport conditions, *no credit is taken for conduction through the neutron shielding material.* Thus, heat transport from the outer intermediate shell surface to the overpack outer shell is conservatively based on heat transfer through the carbon steel radial channel legs alone. Thermal conductivity of the parallel neutron shield and radial channel leg region is given by the following formula:

$$K_{nc} = \frac{K_R\, N_R\, t_R \ln\left[\frac{r_B}{r_A}\right]}{2\pi L_R} + \frac{K_{ns}\, N_R\, t_{ns} \ln\left[\frac{r_B}{r_A}\right]}{2\pi L_R}$$

where (in consistent U.S. units):

K_{ne}	=	effective thermal conductivity of neutron shield region
r_A	=	inner radius of neutron shielding
r_B	=	outer radius of neutron shielding
K_R	=	effective thermal conductivity of carbon steel radial channel leg
N_R	=	total number of radial channel legs (also equal number of neutron shield sections)
t_R	=	minimum (nominal) thickness of each radial channel leg
L_R	=	effective radial heat transport length through radial channel leg
K_{ns}	=	neutron shield thermal conductivity
t_{ns}	=	neutron shield circumferential thickness (between two radial channel legs)

The radial channel leg to outer intermediate shell surface weld thickness is equal to half the plate thickness. The additional weld resistance is accounted for by reducing the plate thickness in the weld region for a short radial span equal to the weld size. Conductivity of the radial carbon steel channel legs based on the full thickness for the entire radial span is correspondingly reduced. Figure 3.4.4 depicts a resistance network developed to combine the neutron shield and radial channel legs resistances to determine an effective conductivity of the neutron shield region. Note that in the resistance network analogy only the annulus region between overpack outer enclosure inner surface and intermediate shells outer surface is considered in this analysis. The effective thermal conductivity of neutron shield region is provided in Table 3.4.8.

3.4.1.1.10 Effective Thermal Conductivity of the Eccentric MPC to Overpack Gap

During horizontal shipment of the HI-STAR System under normal transport conditions, the MPC will rest on the inside surface of the overpack. In the region of line contact, the resistance to heat transfer across the gap will be negligibly small due to a vanishingly small gap thickness. The resistance to heat transfer at other regions along the periphery of the MPC will, however, increase in direct proportion to the thickness of the local gap. This variation in gap thickness can be accounted for in the thermal model by developing a relation for the total heat transferred across the gap as given below:

$$Q_E = 2\int_0^{\pi} \frac{K_{He}}{g(\theta)} L\, R_o\, \Delta T\, d\theta$$

where:

Q_E	=	total heat transfer across the gap (Btu/hr)
K_{He}	=	helium conductivity Btu/ft-hr-°F
L	=	length of MPC (ft.)
R_o	=	MPC radius (ft.)
θ	=	angle from point of line contact
$g(\theta)$	=	variation of gap thickness with angle (ft.)
ΔT	=	temperature difference across the gap (°F)

A corresponding relationship for heat transferred across a uniform gap is given by:

$$Q_c = \frac{K_{eff}}{(R_I - R_O)} 2\pi\, R_O\, L\, \Delta T$$

where R_I is the inside radius of the overpack and K_{eff} is the effective thermal conductivity of an equivalent concentric MPC/overpack gap configuration. From these two relationships, the ratio of effective gap conductivity to helium thermal conductivity in the MPC/overpack region is shown below:

$$\frac{K_{eff}}{K_{He}} = \frac{R_I - R_O}{\pi} \int_0^{\pi} \frac{1}{g(\theta)} d\theta$$

Based on an analysis of the geometry of a thin gap between two eccentrically positioned cylinders, the following relationship is developed for variation of the gap thickness with position:

$$g(\theta)=(R_1-R_0)(1-\cos\theta)+\varepsilon\cos\theta$$

The above equation conservatively accounts for imperfect contact by postulating a minimum gap ε at the point where the two surfaces would ideally form a line of perfect contact. The relatively thin MPC shell is far more flexible than the much thicker overpack inner shell, and will ovalize to yield greater than line contact. The substantial weight of the fuel basket and contained fuel assemblies will also cause the MPC shell to conform to the overpack inner shell. An evaluation based on contact along a line would therefore be reasonable and conservative. However, a minimum gap is assumed to further increase conservatism in this calculation.

Based on an applied gap of 0.02-inch, which is conservative compared to contact along a line, the effective gap thermal conductivity determined from analytical integration [3.4.7] is in excess of 200% of the conductivity of helium gas. In the HI-STAR analysis, a conservative effective gap conductivity equal to twice the helium gas conductivity is applied to the performance evaluation.

3.4.1.1.11 Effective Thermal Conductivity of MPC Basket-to-Shell Aluminum Heat Conduction Elements

The HI-STAR MPCs feature an option to install full-length heat conduction elements fabricated from aluminum alloy 1100 in the large MPC basket-to-shell gaps. Due to the high aluminum alloy 1100 thermal conductivity (about 15 times that of Alloy X), a significant rate of net heat transfer is possible along the thin plates. For conservatism, heat dissipation by the Aluminum Heat Conduction Elements (AHCEs) is ignored in normal transport analyses. This overstates the initial fuel temperature for hypothetical fire accident evaluation. To conservatively compute heating of MPC contents in a hypothetical fire condition, the presence of heat conduction elements in AHCE equipped MPCs is duly recognized.

Figure 3.4.12 shows a mathematical idealization of a heat conduction element inserted between basket periphery panels and the MPC shell. The aluminum insert is shown to cover the MPC basket Alloy X peripheral panel and MPC shell surfaces (Regions I and III depicted in Figure 3.4.12) along the full-length of the basket. Heat transport to and from the aluminum insert is conservatively postulated to occur across a thin helium gap as shown in the figure (i.e., no credit is considered for aluminum insert to Alloy X metal-to-metal contact). Aluminum surfaces inside the hollow region are sandblasted prior to fabrication to result in a rough surface finish which has a significantly higher emissivity compared to smooth surfaces of rolled aluminum. The untreated aluminum surfaces directly facing Alloy X panels have a smooth finish to minimize contact resistance.

Net heat transfer resistance from the hot basket periphery panel to the relatively cooler MPC shell along the aluminum heat conduction element pathway is a sum of three individual resistances in regions labeled I, II, and III. In Region I, heat is transported from the basket to the aluminum insert surface directly facing the basket panel across a thin helium resistance gap. Longitudinal transport of heat (in the z direction) in the aluminum plate (in Region I) will result in an axially non-uniform temperature distribution. Longitudinal one-dimensional heat transfer in the Region I aluminum plate is analytically formulated to result in the following ordinary differential equation for the non-uniform temperature distribution:

$$t\,K_{Al}\frac{\partial^2 T}{\partial z^2} = -\frac{K_{He}}{h}(T_h - T) \qquad \text{(Equation a)}$$

Boundary Conditions

$$\frac{\partial T}{\partial z} = 0 \text{ at } z = 0$$
$$T = T_h' \text{ at } z = P \qquad \text{(Equation b)}$$

where (see Figure 3.4.12):

$T(z)$	=	non-uniform aluminum metal temperature distribution
t	=	conduction element thickness
K_{Al}	=	conduction element conductivity
K_{He}	=	helium conductivity
h	=	helium gap thickness
T_h	=	hot basket temperature
T_h'	=	conduction element Region I boundary temperature at z = P
P	=	conduction element Region I length

Solution of this ordinary differential equation subject to the imposed boundary condition is:

$$(T_h - T) = (T_h - T_h')\left[\frac{e^{\frac{z}{\sqrt{\alpha}}} + e^{-\frac{z}{\sqrt{\alpha}}}}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}}\right] \qquad \text{(Equation c)}$$

where α is a dimensional parameter equal to htK_{Al}/K_{He}. The net heat transfer (Q_I) across the Region I helium gap can be determined by the following integrated heat flux to a conduction element of length L as:

$$Q_I = \int_0^P \frac{K_{He}}{h}(T_h - T)(L)\,dz \qquad \text{(Equation d)}$$

Substituting the analytical temperature distribution result obtained in Equation c into Equation d and then integrating, the following expression for net heat transfer is obtained:

$$Q_I = \frac{K_{He} L \sqrt{\alpha}}{h}\left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}}\right)(T_h - T_h') \qquad \text{(Equation e)}$$

Based on this result, an expression for Region I resistance is obtained as shown below:

$$R_I = \frac{T_h - T_h'}{Q_I} = \frac{h}{K_{He} L \sqrt{\alpha}}\left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}}\right)^{-1} \qquad \text{(Equation f)}$$

Similarly, a Region III resistance expression can be analytically determined as shown below:

$$R_{III} = \frac{(T_c' - T_c)}{Q_{III}} = \frac{h}{K_{He} L \sqrt{\alpha}}\left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}}\right)^{-1} \qquad \text{(Equation g)}$$

A Region II resistance expression can be developed from the following net heat transfer equation in the vertical leg of the conduction element as shown below:

$$Q_{II} = \frac{K_{Al} L t}{W}(T_h' - T_c') \qquad \text{(Equation h)}$$

Hence,

$$R_{II} = \frac{T_h' - T_c'}{Q_{II}} = \frac{W}{K_{Al} L t} \qquad \text{(Equation i)}$$

This completes the analysis for the total thermal resistance attributable to the heat conduction elements equal to sum of the three individual resistances. The total resistance is smeared across the basket-to-MPC shell region as an effective uniform annular gap conductivity (see Figure 3.4.2). Note that heat transport along the conduction elements is an independent conduction path in parallel with conduction and radiation mechanisms in the large helium gaps. Helium conduction and radiation between the MPC basket and the MPC shell is accounted for separately in the ANSYS MPC models described earlier in this section. Therefore, the total MPC basket-to-MPC shell peripheral gaps conductivity will be the sum of the conduction elements effective conductivity and the helium conduction-radiation gap effective conductivity.

3.4.1.1.12 FLUENT Model for HI-STAR Temperature Field Computation

In the preceding subsections, the series of analytical and numerical models to define the thermal characteristics of the various elements of the HI-STAR System are presented. The thermal modeling begins with the replacement of the SNF cross section and surrounding fuel cell space by a solid lamina with an equivalent conductivity. Since radiation is an important constituent of the heat transfer process in the SNF/storage cell space and the rate of radiation heat transfer is a strong function of the surface temperatures, it is necessary to treat the equivalent lamina conductivity as a function of temperature. In fact, because of the relatively large range of temperatures which will exist in a loaded HI-STAR System under the design basis heat loads, it is necessary to include the effect of variation in the thermal conductivity of materials with temperature throughout the system finite volume model. The presence of significant radiation effect in the storage cell spaces adds to the imperative to treat the equivalent lamina conductivity as temperature-dependent.

FLUENT finite volume simulations have been performed to establish the equivalent thermal conductivity as a function of temperature for the limiting (thermally most resistive) BWR and PWR spent fuel types. By utilizing the most limiting SNF (established through a simplified analytical process for comparing conductivities) the numerical idealization for the fuel space conductivity is ensured to be conservative for all non-limiting fuel types.

Having replaced the interior of the cell spaces by solid prismatic (square) columns possessing a temperature-dependent conductivity essentially renders the basket into a non-homogeneous three-dimensional solid where the non-homogeneity is introduced by the honeycomb basket structure. The basket panels themselves are a composite of Alloy X cell wall, Boral neutron absorber, and Alloy X sheathing metal. A conservative approach to replace this composite section with an equivalent "solid wall" is described in a preceding subsection.

In the next step, a planar section of the MPC is considered. The MPC, externally radially symmetric, contains a non-symmetric basket lamina wherein the equivalent fuel space solid squares are separated by the "equivalent" solid metal walls. The space between the basket and the MPC, called the peripheral gap, is filled with helium gas and optionally aluminum heat conduction elements. The equivalent thermal conductivity of this MPC section is computed using a finite element procedure on ANSYS, as described previously. For hypothetical fire conditions the "helium-conduction-radiation" based peripheral gap conductivity and the effective conductivity of aluminum conduction elements are added to obtain a combined effective conductivity. At this stage in the thermal analysis, the SNF/basket/MPC assemblage has been replaced with a two-zone (Figure 3.4.2) cylindrical solid whose thermal conductivity is a strong function of temperature.

The idealization for the overpack is considerably more straightforward. The overpack is radially symmetric except for the Holtite region (discussed in Subsection 3.4.1.1.9). The procedure to replace the multiple shell layers, Holtite-A and radial connectors with an equivalent solid utilizes classical heat conduction analogies, as described in the preceding subsections.

In the final step of the analysis, the equivalent two-zone MPC cylinder, the equivalent overpack shell, the top and bottom plates, and the impact limiters are assembled into a comprehensive finite volume model. A cross section of this axisymmetric model implemented on FLUENT is shown in Figure 3.4.14. A summary of the essential features of this model is presented in the following:

- The overpack shell is represented by 840×9 elements. The effective thermal conductivity of the overpack shell elements is set down as a function of temperature based on the analyses described earlier.

- The overpack bottom plate and bolted closure plate are modeled by 312×9 axisymmetric elements.

- The two-zone MPC "solid" is represented by 1,144×9 axisymmetric elements.

- The space between the MPC "solid" and the overpack interior space is assumed to contain helium.

- Heat input due to insolation is applied to the impact limiter surfaces and the cylindrical surface of the overpack.

- The heat generation in the MPC solid basket region is assumed to be uniform in each horizontal plane, but to vary in the axial direction to correspond to the axial burnup distribution in the active fuel region postulated in Chapter 1.

The finite volume model constructed in this manner will produce an axisymmetric temperature distribution. The peak temperature will occur near the centerline and is expected to correspond to the axial location of peak heat generation. As is shown later, the results from the finite element solution bear out these observations.

3.4.1.1.13 Effect of Fuel Cladding Crud Resistance

In this subsection, a conservatively bounding estimate of the temperature drop across a crud film adhering to a fuel rod during dry storage conditions is determined. The evaluation is performed for a

BWR fuel assembly based on an upper bound crud thickness obtained from PNL-4835 report ([3.3.5], Table 3). The crud present on fuel assemblies is predominantly iron oxide mixed, with small quantities of other metals such as cobalt, nickel, chromium, etc. Consequently, the effective conductivity of the crud mixture is expected to be in the range of typical metal alloys. Metals have thermal conductivities several orders of magnitude larger than that of helium. In the interest of extreme conservatism, however, a film of helium with the same thickness replaces the crud layer. The calculation is performed in two steps. In the first step, a crud film resistance is determined based on bounding maximum crud layer thickness replaced as a helium film on the fuel rod surfaces. This is followed by a peak local cladding heat flux calculation for the smaller GE 7×7 fuel assembly postulated to emit a conservatively bounding decay heat equal to 0.5kW. The temperature drop across the crud film obtained as a product of the heat flux and crud resistance terms is determined to be less than 0.1°F. The calculations are presented below:

Bounding Crud Thickness (δ) (PNL-4835)	=	130μm (4.26×10^{-4} ft)
Crud Conductivity (K)	=	0.1 Btu/ft-hr-°F (conservatively assumed as helium)

GE 7×7 Fuel Assembly:

Rod O.D.	=	0.563"
Active Fuel Length	=	150"
Heat Transfer Area	=	$(7\times7)(\pi\times0.563)\times150/144$
	=	90.3 ft^2
Axial Peaking Factor	=	1.195 (Burnup distribution Table 1.2.15)
Decay Heat	=	500W (conservative assumption)

$$\text{Crud Resistance} = \frac{\delta}{K} = \frac{4.26\times10^{-4}}{0.1} = 4.26\times10^{-3}\,\frac{\text{ft}^2\text{-hr-}^\circ\text{F}}{\text{Btu}}$$

$$\text{Peak Heat Flux} = \frac{(500\times3.417)\ \text{Btu/hr}}{90.3\ \text{ft}^2}\times1.195$$

$$= 18.92\times1.195 = 22.6\,\frac{\text{Btu}}{\text{ft}^2\text{-hr}}$$

Temperature drop (ΔT_c) across crud film:

$$= 4.26\times10^{-3}\,\frac{\text{ft}^2\text{-hr-}^\circ\text{F}}{\text{Btu}}\times22.6\,\frac{\text{Btu}}{\text{ft}^2\text{-hr}}$$

$$= 0.096^\circ\text{F}$$

$$(\text{i.e., less than } 0.1^\circ\text{F})$$

Therefore, it is concluded that deposition of crud does not materially change the SNF cladding temperature.

3.4.1.1.14 Maximum Time Limit During Wet Transfer

While loading an empty HI-STAR System for transport directly from a spent fuel pool, water inside the MPC cavity is not permitted to boil. Consequently, uncontrolled pressures in the de-watering, purging, and recharging system that may result from two-phase condition, are completely avoided. This requirement is accomplished by imposing a limit on the maximum allowable time duration for fuel to be submerged in water after a loaded HI-STAR cask is removed from the pool and prior to the start of vacuum drying operations.

When the HI-STAR overpack and the loaded MPC under water-flooded conditions are removed from the pool, the combined mass of the water, the fuel, the MPC, and the overpack will absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the entire system with time, starting from an initial temperature of the contents. The rate of temperature rise is limited by the thermal inertia of the HI-STAR system. To enable a bounding heat-up rate determination for the HI-STAR system, the following conservative assumptions are imposed:

i. Heat loss by natural convection and radiation from the exposed HI-STAR surfaces to the pool building ambient air is neglected (i.e., an adiabatic temperature rise calculation is performed).

ii. Design Basis maximum decay heat input from the loaded fuel assemblies is imposed on the HI-STAR system.

iii. The smallest of the *minimum* MPC cavity-free volumes between the two MPC types is considered for flooded water mass determination.

iv. Fifty percent of the water mass in the MPC cavity is credited towards water thermal inertia evaluation.

Table 3.4.19 summarizes the weights and thermal inertias of several components in the loaded HI-STAR system. The rate of temperature rise of the HI-STAR and its contents during an adiabatic heat-up is governed by the following equation:

$$\frac{dT}{d\tau} = \frac{Q}{C_h}$$

where:

Q = decay heat load (Btu/hr) [equal to Design Basis maximum (between the two MPC types) 20.0 kW (i.e., 68,260 Btu/hr)]

C_h = combined thermal inertia of the loaded HI-STAR system (Btu/°F)

T = temperature of the contents (°F)

τ = time after HI-STAR system is removed from the pool (hr)

A bounding heat-up rate for the HI-STAR system contents is determined to be equal to 2.19°F/hr. From this adiabatic rate of temperature rise estimate, the maximum allowable time duration (t_{max}) for fuel to be submerged in water is determined as follows:

$$t_{max} = \frac{T_{boil} - T_{initial}}{dT/d\tau}$$

where:

T_{boil} = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)

$T_{initial}$ = initial temperature of the HI-STAR contents when removed from the pool

Table 3.4.20 provides a summary of t_{max} at several initial HI-STAR contents temperatures.

As set forth in Section 7.4, in the unlikely event where the maximum allowable time provided in Table 3.4.20 is found to be insufficient to complete all wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water will enter via the MPC lid drain port connection and heated water will exit from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

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

where:

M_W = minimum water flow rate (lb/hr)

C_{pw} = water heat capacity (Btu/lb-°F)

T_{max} = maximum MPC cavity water mass temperature

T_{in} = temperature of water supply to MPC

With the MPC cavity water temperature limited to 150°F, MPC inlet water maximum temperature equal to 125°F and at the design basis maximum heat load, the water flow rate is determined to be 2,731 lb/hr (5.5 gpm).

3.4.1.1.15 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

Before a loaded HI-STAR System can be unloaded (i.e., fuel removed from the MPC) the cask must be cooled from the operating temperatures and reflooded with water*. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. However, the extremely rapid cooldown rates that are typical during water injection, to which the hot cask internals and fuel cladding are subjected to, may result in uncontrolled thermal stresses and failure in the structural members. Moreover, water injection results in large amounts of steam generation and unpredictable transient two-phase flow conditions inside the MPC cavity, which may result in over-pressurization of the MPC helium retention boundary and a potentially unacceptable reduction in the safety margins to prevent criticality. To avoid potential safety concerns related to rapid cask cooldown by direct water quenching, the HI-STAR MPCs are designed to be cooled in a gradual manner, thereby eliminating thermal shock loads on the cask internals and fuel cladding.

In the unlikely event that a HI-STAR system is required to be unloaded, it will be transported back to the fuel handling building. Prior to reflooding the MPC cavity with water, a forced flow helium recirculation system with adequate flow capacity shall be operated to remove the decay heat and initiate a slow cask cooldown lasting for several days. The operating procedures in Section 7.2 provide a detailed description of the steps involved in the cask unloading. In this section, an analytical evaluation is presented to provide the basis for helium flow rates and time of forced cooling to meet the objective of eliminating thermal shock when the MPC cavity is eventually flooded with water.

Under a closed loop forced helium circulation condition, the helium gas is cooled via an external chiller, down to 100°F, and then introduced inside the MPC cavity from the drain line near the bottom baseplate. The helium gas enters the MPC basket from the bottom oversized flow holes and moves upward through the hot fuel assemblies, removing heat and cooling the MPC internals. The heated helium gas exits from the basket top and collects in the top plenum, from where it is expelled through the MPC lid vent connection to the helium recirculation and cooling system. The bulk average temperature reduction of the MPC contents as a function of time is principally dependent upon the rate of helium circulation. The temperature transient is governed by the following heat balance equation:

$$C_h \frac{dT}{d\tau} = Q_D - m\, C_p (T - T_i) - Q_c$$

Initial Condition: $T = T_0$ at $\tau = 0$

---

* Certain fuel configurations in PWR MPCs require Borated water for criticality control (Chapter 6). Such MPCs are reflooded with Borated water.

where:

T = MPC bulk average temperature (°F)

T_o = initial MPC bulk average temperature in the HI-STAR system (483°F*)

τ = time after start of forced circulation (hr)

Q_D = decay heat load (Btu/hr) (equal to Design Basis maximum 20.0 kW (i.e., 68,260 Btu/hr))

m = helium circulation rate (lb/hr)

C_p = helium heat capacity (Btu/lb-°F) (equal to 1.24 Btu/lb-°F)

Q_c = heat rejection from cask exposed surfaces to ambient (Btu/hr) (conservatively neglected)

C_h = thermal capacity of the loaded MPC (Btu/°F) (For a bounding upper bound 100,000 lb loaded MPC weight, and heat capacity of Alloy X equal to 0.12 Btu/lb-°F, the heat capacity is equal to 12,000 Btu/°F)

T_i = MPC helium inlet temperature (°F)

The differential equation is analytically solved, yielding the following expression for time-dependent MPC bulk temperature:

$$T(t)=(T_i+\frac{Q_D}{m\,C_p})(1-e^{\frac{mC_p}{C_h}t})+T_o e^{\frac{mC_p}{C_h}t}$$

This equation is used to determine the minimum helium mass flow rate that would cool the MPC cavity down from initially hot conditions to less than 200°F. For example, to cool the MPC to less than 200°F in 72 hours would required a helium mass flow rate of 574 lb/hr (i.e., 859 SCFM).

Once the helium gas circulation has cooled the MPC internals to less than 200°F, water can be injected to the MPC without risk of boiling and the associated thermal stress concerns. Because of the relatively long cooldown period, the thermal stress contribution to the total cladding stress would be negligible, and the total stress would therefore be bounded by the normal (dry) condition. The

* Bounding for HI-STAR normal transport.

elimination of boiling eliminates any concern of over-pressurization due to steam production.

3.4.1.1.16 MPC Evaluation Under Drying Conditions

The initial loading of SNF in the MPC requires that the water within the MPC be drained, residual moisture removed and MPC filled with helium. This operation on the HI-STAR MPCs will be carried out using a Forced Helium Dehydrator (FHD) for a "load-and-go" operation. A "load-and-go" operation is defined as an activity wherein an MPC is loaded for direct off-site shipment in a HI-STAR transport cask. MPCs prepared via other competent methods for MPC drying as approved by the NRC on other dockets (1008 and 1014) are duly recognized for transport under this docket.

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a closed loop system consisting of a condenser, a demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. Appendix 3.B contains detailed discussion of the design and operation criteria for the FHD system.

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

3.4.1.1.17 Effects of Helium Dilution from Fuel Rod Gases

In this subsection, the generic cask transportation accident issue raised in a USNRC Spent Fuel Project Office (SFPO) staff guidance letter† is addressed. This issue directs cask designers to evaluate the impact of fission gas release into the canister, from a 100% fuel rods rupture accident, on the cask component temperatures and pressures when the MNOP* is within 10% of the design pressure. To determine whether the HI-STAR System falls within the stipulated criteria, the MNOP results from Table 3.4.15 are provided below:

† SFPO Director's Interim Staff Guidance Letter(s), W.F. Kane, (Interim Staff-Guidance-7), October 8, 1998.

* MNOP is a regulatory term defined in NUREG-1617 as the maximum gauge pressure that would develop in the containment in a period of 1 year under the heat condition specified in 10 CFR 71.71(c)(1) in the absence of venting, external ancillary cooling or operational controls.

Canister	MNOP (psig)	Threshold Criteria* for Accident Evaluation (psig)
MPC-24	88.8	90
MPC-68	86.9	90
MPC-24E	88.9	90
MPC-32	89.3	90

As shown above the MNOPs are below the threshold and an accident evaluation is not required. Nevertheless, for illustrative purposes, a 100% rods rupture accident for a HI-STAR package with an MPC-24 canister is evaluated..
Under a severe hypothetical accident scenario 100% of the fuel rods may rupture, releasing the rod fill gas (helium) and a portion of the gaseous fission products (^3H, ^{85}Kr, ^{129}I and ^{131}Xe). The gaseous fission products release fractions are stipulated in NUREG-1536. The released gases will mix with the MPC backfill gas and reduce its thermal conductivity. This reduction in conductivity will result in a small increase in MPC temperatures and pressures.

Appendix C of NUREG/CR-0497 [3.4.13] describes a method for calculating the effective thermal conductivity of a mixture of gases. The same method is also described by Rohsenow and Hartnett [3.2.2]. The following expression is provided by both references:

$$k_{mix} = \sum_{i=1}^{n} \left(\frac{k_i x_i}{x_i + \sum_{\substack{j=1 \\ j \neq i}}^{n} \varphi_{ij} x_j} \right)$$

where:

k_{mix} = thermal conductivity of the gas mixture (Btu/hr-ft-°F)
n = number of gases
k_i = thermal conductivity of gas component i (Btu/hr-ft-°F)
x_i = mole fraction of gas component i

In the preceding equation, the term φ_{ij} is given by the following:

$$\varphi_{ij} = \phi_{ij} \left[1 + 2.41 \frac{(M_i - M_j)(M_i - 0.142 \cdot M_j)}{(M_i + M_j)^2} \right]$$

where M_i and M_j are the molecular weights of gas components i and j, and ϕ_{ij} is:

* Accident evaluation required when MNOP is within 10% of the design pressure. This translates to a pressure that is between 100 psig (HI-STAR Design Pressure (Table 2.1.1) and 90 psig.

$$\phi_{ij} = \frac{\left[1+\left(\frac{k_i}{k_j}\right)^{\frac{1}{2}}\left(\frac{M_i}{M_j}\right)^{\frac{1}{4}}\right]^2}{2^{\frac{3}{2}}\left(1+\frac{M_i}{M_j}\right)^{\frac{1}{2}}}$$

Table 3.4.30 presents a summary of the gas mixture thermal conductivity calculations for an MPC-24 containing design basis PWR fuel assemblies.

Having calculated the gas mixture thermal conductivity, the effective thermal conductivity of the design basis PWR fuel assembly is calculated using the finite-volume model described in Subsection 3.4.1.1.2. Only the helium gas conductivity is changed, all other modeling assumptions are the same. The fuel assembly effective thermal conductivity with diluted helium is compared to that with undiluted helium in Table 3.4.31.

Next, the effective thermal conductivities of the MPC fuel basket and basket periphery regions are determined as described in Subsections 3.4.1.1.3 and 3.4.1.1.4. This calculation incorporates both the diluted helium thermal conductivity and the effective thermal conductivity of the fuel assembly with diluted helium. ~~The Rayleigh effect thermal conductivity multipliers are unchanged in this analysis. This is conservative because the released rod gases will increase the average fluid density and decrease the gas thermal conductivity, consequently increasing the Rayleigh number.~~ *In the evaluation of helium dilution by high molecular weight gases (fission gas releases from hypothetical rupture of fuel rods) the increase in convection heat transfer in the basket peripheral spaces due to a substantial rise in gas density is recognized.* The effective thermal conductivities with diluted helium are compared to those with undiluted helium in Table 3.4.31.

The MPC fuel basket effective thermal conductivities are input to a finite-volume model of the HI-STAR System arranged for transport. The cask system temperature distribution with diluted MPC helium is determined using the finite-volume model, as described in Subsection 3.4.1.1.12. Design basis normal environmental conditions are applied to the model and a temperature field solution obtained. Cask system temperatures with diluted MPC helium are summarized in Table 3.4.32.

The slightly higher MPC cavity temperature with MPC helium dilution will result in a small perturbation in MPC internal pressure. Based on the temperature field obtained with helium dilution, the MPC internal pressure is determined using the Ideal Gas Law. The calculated MPC internal pressure with helium dilution is presented in Table 3.4.33.

The results of analyses presented in this subsection are performed to illustrate the effect of a hypothetical 100% rods rupture on a HI-STAR package with an MPC-24. . Even under the extreme postulated conditions, the MPC component temperatures and pressures remain substantially below the design limits. .

3.4.1.1.18 <u>HI-STAR Temperature Field With Low Heat Emitting Fuel</u>

The HI-STAR 100 thermal evaluations for BWR fuel are divided in two groups of fuel assemblies proposed for storage in MPC-68. These groups are classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE group of fuel assemblies are characterized by low burnup, long cooling time, and short active fuel lengths. Consequently, their heat loads are dwarfed by the DB group of fuel assemblies. The Dresden-1 (6x6 and 8x8), *Quad*$^+$, and Humboldt Bay (7x7 and 6x6) fuel characteristics warrant their classification as LHE fuel. These characteristics, including burnup and cooling time limits imposed on this class of fuel, are presented in Table *1.2.23* ~~2.1.6~~. This fuel (*except Quad*$^+$ is permitted to be loaded when encased in Damaged Fuel Containers (DFCs). As a result of interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary, this loading configuration is bounding for thermal evaluation. In Subsection 3.4.1.1.2, two canister designs for encasing LHE fuel are evaluated – a previously approved Holtec Design (Holtec Drawing-1783) and an existing canister in which some of the Dresden-1 fuel is currently stored (Transnuclear D-1 Canister). The most resistive fuel assembly determined by analytical evaluation is considered for thermal evaluation (see Table *3.4.5* ~~4.4.6~~). The MPC-68 basket effective conductivity, loaded with the most resistive fuel assembly from the LHE group of fuel (*encased in a canister*) is provided in Table *3.4.6* ~~4.4.7~~. To this basket, LHE fuel decay heat load, is applied and a HI-STAR 100 System temperature field obtained. The low heat load burden limits the initial peak cladding temperature to less than 579°F which is substantially below the cladding temperature limit (Table 3.3.1).

A thoria rod canister designed to hold a maximum of 20 fuel rods arrayed in a 5x4 configuration is currently stored at the Dresden-1 spent fuel pool. The fuel rods contain a mixture of enriched UO_2 and Thorium Oxide in the fuel pellets. The fuel rods were originally constituted as part of an 8x8 fuel assembly and used in the second and third cycle of Dresden-1 operation. The maximum fuel burnup of these rods is quite low (~13,100 MWD/MTU). The thoria rod canister internal design is a honeycomb structure formed from 12 gage stainless steel plates. The rods are loaded in individual square cells and are isolated from each other by the cell walls. The few number of rods (18 per assembly) and very low burnup of fuel stored in these Dresden-1 canisters render them as miniscule sources of decay heat. The canister all-metal internal honeycomb construction serves as an additional means of heat dissipation in the fuel cell space. In accordance with preferential fuel loading requirements, low burnup fuel shall be loaded toward the basket periphery (i.e., away from the hot central core of the fuel basket). All these considerations provide ample assurance that these fuel rods will be stored in a benign thermal environment and therefore remain protected during transport.

3.4.1.2 <u>Test Model</u>

A detailed analytical model for evaluating the thermal design of the HI-STAR System was developed using the FLUENT CFD code and the industry standard ANSYS modeling system as discussed in Subsection 3.4.1.1. Furthermore, the analysis incorporates many conservative assumptions in order to demonstrate compliance with specified temperature limits for operation with adequate margins. In view of these considerations, the HI-STAR thermal design complies with the thermal criteria set forth in the design basis for normal transport conditions. Additional experimental verification of the thermal design is therefore not required. Acceptance and periodic thermal testing for the HI-STAR System is discussed in Sections 8.1 and 8.2.

3.4.2 Maximum Temperatures Under Normal Transport Conditions

Both MPC-basket designs developed for the HI-STAR System have been analyzed to determine temperature distributions under normal transport conditions. In the HI-STAR System thermal analysis models developed on FLUENT, the overpack impact limiters are included in the finite volume geometry. However, no credit is considered for the presence of heat conducting aluminum honeycomb material. In other words, heat transmission through the ends is conservatively neglected in the analysis. The thermal results are therefore bounding with respect to impact limiter design. The MPC baskets are considered to be loaded at design-basis maximum heat load with PWR or BWR fuel assemblies, as appropriate.

As discussed in Subsection 3.4.1.1.1, the thermal analysis is performed using a submodeling process where the results of an analysis on an individual component are incorporated into the analysis of a larger set of components. Specifically, the submodeling process yields directly computed fuel temperatures from which fuel basket temperatures are indirectly calculated. This modeling process differs from previous analytical approaches wherein the basket temperatures were evaluated first and then a basket-to-cladding temperature difference calculation by Wooten-Epstein or other means provided a basis for cladding temperatures. Subsection 3.4.1.1.2 describes the calculation of an effective fuel assembly thermal conductivity for an equivalent homogenous region. It is important to note that the result of this analysis is a function for thermal conductivity versus temperature. This function for fuel thermal conductivity is then input to the fuel basket effective thermal conductivity calculation described in Subsection 3.4.1.1.4. This calculation uses a finite-element methodology, wherein each fuel cell region containing multiple finite-elements has temperature varying thermal conductivity properties. The resultant temperature varying fuel basket thermal conductivity computed by this basket-fuel composite model is then input to the fuel basket region of the FLUENT cask model.

Because the FLUENT cask model incorporates the results of the fuel basket submodel, which in turn incorporates the fuel assembly submodel, the peak temperature reported from the FLUENT model is the peak temperature in any component. In a dry storage cask, the hottest components are the fuel assemblies. It should be noted that, because the fuel assembly models described in Subsection 3.4.1.1.2 include the fuel pellets, the FLUENT calculated peak temperatures reported in Tables 3.4.10 and 3.4.11 are actually peak pellet centerline temperatures which bound the peak cladding temperatures. We conservatively assume that the peak clad temperature is equal to the peak pellet centerline temperature.

From a thermal/hydraulic standpoint, the HI-STAR transport cask must cover two scenarios:

i. MPCs equipped with AHCEs
ii. MPCs without AHCEs

In the thermal analysis submitted in support of HI-STAR's original transport certification, which we now refer to as the Baseline Thermal Model (BTM), the AHCEs are included in the thermal models and the basket thermal model is constructed in an exceedingly conservative manner. In particular, the axial conductance of the basket fuel assemblage is assumed to be equal to the in-plane conductance (in reality, the in-plane conductance is much smaller than the axial conductance due to the presence of physical gaps between the fuel and the cell and within the fuel assemblies). For the Scenario (ii) analysis, such an overarching conservatism is removed while certain other less sweeping conservatisms are retained. The revised model, which we refer to as the Refined Thermal Model (RTM), forms the licensing basis for thermal evaluation. The conservatisms germane to the RTM are summarized in Appendix 3.A. To summarize, the principal difference between the BTM and RTM are as follows:

Item	*Description*	*BTM Assumption*	*RTM Assumption*
1	AHCE heat dissipation	Included	Excluded
2	Rayleigh effect	Included	Excluded
3	Basket Axial Conductivity	*Grossly Understated*	Realistic modeling of axial conductivity (See discussion in Subsection 3.4.1.1.4)

For representative PWR (MPC-24) and BWR (MPC-68) MPC-basket configurations with AHCEs installed, the temperature contours obtained with the Baseline Thermal Model (BTM) corresponding to *steady-state hot* conditions (100°F ambient, maximum design basis maximum decay heat and full insolation) are shown in Figures 3.4.16 and 3.4.17. Figures 3.4.19 and 3.4.20 show the axial temperature variation of the hottest fuel rod in the MPC-24 and MPC-68 basket designs, respectively. Figures 3.4.22 and 3.4.23 show the radial temperature profile in the MPC-24 and MPC-68 basket designs, respectively, in the horizontal plane where maximum fuel cladding temperature is indicated. Tables 3.4.10 and 3.4.11 summarize maximum calculated temperatures in different parts of the HI-STAR System at design-basis maximum decay heat loads. Tables 3.4.28 and 3.4.29 summarize the peak fuel cladding temperatures with heat loads lower than the design basis maximum. In Tables 3.4.22 and 3.4.23, maximum calculated temperatures in different parts of the HI-STAR System under *steady-state cold* conditions (-40°F ambient, maximum design basis maximum decay heat and no insolation) are summarized. To confirm the BTM fuel temperatures provided herein are bounding for all MPCs without the AHCEs option (MPC-24, MPC-24E, MPC-32 and MPC-68) a Refined Thermal Model (RTM) is articulated as discussed in the preceding paragraph. As shown next, the results of the refined calculations confirm the BTM results are bounding.

Maximum Cladding Temperatures		
MPC Type	BTM [°F]	RTM [°F]
PWR	701	671 (MPC-24) 668 (MPC-24E) 699 (MPC-32)
BWR	713	642 (MPC-68)

The following additional observations can be derived by inspecting the temperature field obtained from the finite element analysis:

- The maximum fuel cladding temperature is well within the PNL recommended temperature limit.
- The maximum temperature of basket structural material is well within the stipulated design temperatures.
- The maximum temperature of the Boral neutron absorber is below the material supplier's recommended limit.
- The maximum temperatures of the MPC helium retention boundary materials are well below their respective ASME Code limits.
- The maximum temperatures of the aluminum heat conduction elements are well below the stipulated design temperature limits.
- The maximum temperature of the HI-STAR containment boundary materials is well below their respective ASME Code limits.
- The neutron shielding material (Holtite-A) will not experience temperatures in excess of its qualified limit.

The above observations lead us to conclude that the temperature field in the HI-STAR System with a fully loaded MPC containing design-basis heat emitting SNF complies with all regulatory and industry thermal requirements for normal conditions of transport. In other words, the thermal environment in the HI-STAR System will be conducive to safe transport of spent nuclear fuel.

3.4.2.1 Maximum Accessible Surface Temperatures

Access to the HI-STAR overpack cylindrical surface is restricted by the use of a personnel barrier (See Holtec Drawing *3930, Sheet 3* ~~1809~~ in Chapter 1, *Section 1.4*). Therefore, the HI-STAR System surfaces accessible during normal transport are the exposed impact limiter surfaces outside the personnel barrier. In this subsection, the exposed impact limiter surface temperatures are computed by including heat transmission from the hot overpack ends through the impact limiters. A conservatively bounding analysis is performed by applying the thermal conductivity of aluminum to the encased aluminum-honeycomb material in the impact limiter shells to the normal condition thermal model discussed earlier in this chapter. In this manner heat transport to the exposed surfaces from the hot overpack is maximized and accessible surface temperatures over estimated. The maximum exposed cask surface temperatures for a PWR MPC (MPC-24) and a BWR MPC (MPC-68) at design maximum heat loads are 142°F and 139°F respectively. In Figure 3.4.28, a color contour map of the regions of HI-STAR System less than 185°F (358°K) is depicted for the hotter MPC-24 basket design. From this map, it is apparent that the accessible (impact limiter) surface temperatures are below the 10CFR71.43(g) mandated limit by a significant margin.

3.4.3 Minimum Temperatures

As specified in 10CFR71, the minimum ambient temperature conditions for the HI-STAR System are -20°F and a cold environment at -40°F. The HI-STAR System design does not have any minimum decay heat load restrictions for transport. Therefore, under zero decay heat load in combination with no solar input conditions, the temperature distribution will be uniformly equal to the imposed minimum ambient conditions. All HI-STAR System materials of construction would satisfactorily perform their intended function in the transport mode at this minimum postulated temperature condition. Evaluations in Chapter 2 demonstrate the acceptable structural performance of the overpack and MPC steel materials at low temperature. Shielding and criticality functions of the HI-STAR System materials (Chapters 5 and 6) are unaffected by exposure to this minimum temperature.

3.4.3.1 Post Rapid Ambient Temperature Drop Overpack Cooldown Event

In this section, the thermal response of the HI-STAR overpack to a rapid ambient temperature drop is analyzed and evaluated. The ambient temperature is postulated to drop from the maximum to minimum temperature under normal condition of transport in a very short time (100°F to –40°F during a 1 hour period) and is assumed to hold steady at –40°F thereafter. The initial overpack condition prior to this rapid temperature drop corresponds to normal steady state transport with maximum design basis heat load. During this postulated cooldown event, the outer surface of the overpack will initially cool more rapidly than the bulk of metal away from the exposed surfaces. Consequently, it is expected that the through-thickness temperature gradients will increase for a period of time, reach a maximum and follow an asymptotic return to the initial steady condition through thickness temperature gradients as the overpack temperature field approaches the –40°F ambient steady condition. The results of the transient analysis reported in this sub-section verify these observations.

Noting that the state of thermal stress is influenced by changes in the overpack temperature field during the cooldown transient, a number of critical locations in the containment boundary depicted in Figure 3.4.24 are identified as pertinent to a structural integrity evaluation discussed in Subsection 2.6.2.3 of this SAR. Locations (1) and (2) are chosen to track the through-thickness temperature gradients in the overpack top forging which is directly exposed to the ambient. Locations (3) and (4) are chosen to track the overpack inner containment shell through-thickness temperature gradient in a plane of maximum heat generation (i.e. active fuel mid-height) where the heat fluxes and corresponding temperature gradients are highest. Locations (A) and (B) are similarly chosen to track the temperature differential in the multi-layered shells (outer-to-inner shells).

The normal transport condition thermal model discussed previously in this chapter is employed in the overpack cooldown transient analysis. This analysis is carried out by applying time-dependent thermal boundary conditions to the model and starting the transient solution in the FLUENT program. In the cooldown event, the ambient temperature is decreased from 100°F to –40°F in 10°F steps every 4 minutes (i.e. a total of 14 steps lasting 56 minutes). The ambient temperature is held constant thereafter. The maximum design basis heat load cask (i.e. the MPC-24 design) was selected to maximize the thermal gradients (by Fourier's Law, thermal gradient is proportional to heat flow). The overpack cooldown event is tracked by the thermal model for a period of 24 hours and results are reported in Figures 3.4.25 through 3.4.27 as discussed below.

In Figure 3.2.25, the overpack containment through-thickness temperature gradient responses are plotted. From this figure, it is evident that the exposed surface of the overpack forging (location (2)) initially cools at a faster rate than the recessed location (1). A similar but less pronounced result is observed in the multi-layered shells temperature changes depicted in Figure 3.4.26. This out-of-phase rate of cooling results in an increasing temperature gradient through the overpack metal layers. The thermal response of deeply recessed locations (3) and (4) show gradual temperature changes that follow each other closely. In other words, while through-thickness temperature gradients in the forging are somewhat altered the overpack inner shell gradients are essentially unchanged during the cooldown period. A closer examination of the forging temperature gradient is therefore warranted.

In Figure 3.4.27, the time dependent forging through thickness temperature differential is depicted. The gradient increases to a maximum in a short time period followed by a slow return towards the starting state. In absolute terms, both the steady state and transient temperature gradients in the forging are quite modest. In the steady state the forging through thickness temperature gradient is approximately 3°F. This value reaches a maximum plateau of 7°F during the transient event (Figure 3.4.27). The incremental thermal stress arising from this short-term gradient elevation is computed and discussed in Subsection 2.6.2.3 of this SAR.

3.4.4 Maximum Internal Pressures

The MPC is initially filled with dry helium after fuel loading and prior to sealing the MPC lid port cover plates and closure ring. During normal transport conditions, the gas temperature within the MPC rises to its maximum operating temperature as determined by the thermal analysis methodology described earlier (see Subsection 3.4.1). The gas pressure inside the MPC will increase with rising temperature. The pressure rise is determined using the Ideal Gas Law which states that the absolute pressure of a fixed volume of entombed gas is proportional to its absolute temperature.

The HI-STAR Maximum Normal Operating Pressure (MNOP) is calculated for10 CFR71.71(c)(1) heat condition (100°F ambient & insolation) and the HI-STAR Overpack passively cooled at design maximum heat load. For other lower than design maximum heat load scenarios, (e.g. transport with Trojan fuel) the MNOP results are confirmed to be bounding. . In Tables 3.4.13 and 3.4.14, summary calculations for determining net free volume in the PWR and BWR canisters are presented. Based on a 30% release of the significant radioactive gases, a 100% release of the rod fill gas from postulated cladding breaches, the net free volume and the initial fill gas pressure (see Table 3.3.2), the MNOP results are given in Table 3.4.15. The overpack containment boundary MNOP for a hypothetical MPC breach condition is bounded by the MPC pressure results reported in this table.

3.4.5 Maximum Thermal Stresses

Thermal expansion induced mechanical stresses due to imposed non-uniform temperature distributions have been determined and reported in Chapter 2. Tables 3.4.17 and 3.4.18 summarize the HI-STAR System components temperatures, under steady-state hot conditions, for structural evaluation.

Additionally, Table 3.4.24 provides a summary of MPC helium retention boundary temperatures during normal transport conditions (steady state hot). Structural evaluations in Section 2.6 reference these temperature results to demonstrate the MPC helium retention boundary integrity.

3.4.6 Evaluation of System Performance for Normal Conditions of Transport

The HI-STAR System thermal analysis is based on detailed and complete heat transfer models that properly account for radiation, conduction and natural convection modes of heat transfer. The thermal models incorporate many conservative assumptions that are listed below. A quantitative evaluation of HI-STAR conservatisms is provided in Appendix 3.A.

1. No credit for gap reduction between the MPC and overpack due to differential thermal expansion under hot condition is considered.

2. No credit is considered for MPC basket internal thermosiphon heat transfer. Under a perfectly horizontal transport condition, axial temperature gradients with peaking at active fuel mid-height induces buoyancy flows from both ends of the basket in each MPC cell. Buoyancy flow in shallow horizontal channels has been widely researched and reported in the technical literature [3.4.10 to 3.4.12]. An additional mode of heat transport due to thermosiphon flow within the basket cells is initiated for any cask orientation other than a perfectly horizontal condition. In practice this is a highly likely scenario. However, in the interest of conservatism, no credit is considered for this mode of heat transfer.

3. An upper bound solar absorbtivity of unity is applied to all exposed surfaces.

4. No credit considered for radiative heat transfer between the Boral neutron absorber panels and the Boral pocket walls, or for the presence of helium in the pocket gaps.

5. No credit is considered for conduction through the neutron shielding materials.

6. No credit is considered for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the MPC basket supports. The fuel assemblies and MPC basket are conservatively considered to be in concentric alignment.

7. No credit considered for presence of highly conducting aluminum honeycomb material inside impact limiters.

8. The fuel assembly contribution to MPC basket axial conductivity is conservatively limited to the fuel cladding only (i.e. axial heat transfer through fuel pellets is neglected).

9. The MPC is assumed to be loaded with the SNF type which has the maximum equivalent thermal resistance of all fuel types in its category (BWR or PWR), as applicable.

10. The design basis maximum decay heat loads are used for all thermal-hydraulic analyses. For casks loaded with fuel assemblies having decay heat generation rates less than design basis, additional thermal margins of safety will exist.

11. Interfacial contact conductance of multi-layered intermediate shell contacting layers was conservatively determined to bound surface finish, contact pressure, and base metal conductivity conditions.

12. Flow turbulation in the MPC space neglected.

Temperature distribution results obtained from a conservatively developed thermal model show that maximum fuel cladding temperature limits are met with adequate margins. Margins during actual normal transport conditions are expected to be greater due to the many conservative assumptions incorporated in the analysis. The maximum local temperatures in the neutron shield and overpack seals are lower than design limits. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep induced degradation. Furthermore, structural evaluation (Chapter 2) has demonstrated that stresses (including those induced due to imposed temperature gradients) are within ASME B&PV Code limits. Section 3.6 provides a discussion of compliance with the regulatory requirements and acceptance criteria listed in Section 3.0. As a result of the above-mentioned considerations, it is concluded that the HI-STAR thermal design is in compliance with 10CFR71 requirements for normal conditions of transport.

Table 3.4.1

CLOSED CAVITY NUSSELT NUMBER*
RESULTS FOR HELIUM FILLED MPC PERIPHERAL VOIDS

	Case (i) Nusselt Number		Case (ii) Nusselt Number	
Temperature (°F)	MPC-24, MPC-24E, MPC-32	MPC-68	MPC-24, MPC-24E, MPC-32	MPC-68
200	6.93	4.72	5.45	3.46
450	5.44	3.71	4.09	2.58
700	4.60	3.13	3.36	2.12

* For conservatism, the heat dissipation enhancement due to Rayleigh effect discussed in Sub-section 3.4.1.1.5 is ignored.

Table 3.4.2

RELATIONSHIP BETWEEN HI-STAR SYSTEM REGIONS AND MATHEMATICAL MODEL DESCRIPTIONS

HI-STAR System Region	Mathematical Model	Subsections
Fuel Assembly	Fuel Region Effective Thermal Conductivity	3.4.1.1.2
MPC	Effective Thermal Conductivity of Boral/Sheathing/Box Wall Sandwich	3.4.1.1.3
	Basket In-Plane Conductive Heat Transport	3.4.1.1.4
	Heat Transfer in MPC Basket Peripheral Region	3.4.1.1.5
	Effective Thermal Conductivity of MPC Basket-to-Shell Aluminum Heat Conduction Elements	3.4.1.1.11
Overpack	Effective Conductivity of Multi-Layered Intermediate Shell Region	3.4.1.1.6
	Effective Thermal Conductivity of Holtite Neutron Shielding Region	3.4.1.1.9
Ambient Environment	Heat Rejection from Overpack Exterior Surfaces	3.4.1.1.7
	Solar Heat Input	3.4.1.1.8
Assembled Cask Model	Overview of the Thermal Model	3.4.1.1.1
	Effective Conductivity of MPC to Overpack Gap	3.4.1.1.10
	FLUENT Model for HI-STAR	3.4.1.1.12

Table 3.4.3

THIS TABLE IS INTENTIONALLY DELETED.

Table 3.4.4

SUMMARY OF PWR FUEL ASSEMBLIES
EFFECTIVE THERMAL CONDUCTIVITIES

No.	Fuel	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
1	W 17×17 OFA	0.182	0.277	**0.402**
2	W 17×17 Std	0.189	0.286	0.413
3	W 17×17 Vantage-5H	0.182	0.277	0.402
4	W 15×15 Std	0.191	0.294	0.430
5	W 14×14 Std	0.182	0.284	0.424
6	W 14×14 OFA	**0.175**	**0.275**	0.413
7	B&W 17×17	0.191	0.289	0.416
8	B&W 15×15	0.195	0.298	0.436
9	CE 16×16	0.183	0.281	0.411
10	CE 14×14	0.189	0.293	0.435
11	HN† 15×15 SS	0.180	0.265	0.370
12	W 14×14 SS	0.170	0.254	0.361
13	B&W 15x15 Mark B-11	0.187	0.289	0.424
14	CE 14x14 (MP2)	0.188	0.293	0.434

Note: Boldface values denote the lowest thermal conductivity in each column (excluding stainless steel clad fuel assemblies).

† Haddam Neck B&W or Westinghouse stainless steel clad fuel assemblies.

Table 3.4.5

SUMMARY OF BWR FUEL ASSEMBLIES EFFECTIVE THERMAL CONDUCTIVITIES

No.	**Fuel**	**@ 200°F (Btu/ft-hr-°F)**	**@ 450°F (Btu/ft-hr-°F)**	**@ 700°F (Btu/ft-hr-°F)**
1	Dresden 1 8×8*	0.119	0.201	0.319
2	Dresden 1 6×6	0.126	0.215	0.345
3	GE 7×7	0.171	0.286	0.449
4	GE 7×7R	0.171	0.286	0.449
5	GE 8×8	0.168	0.278	0.433
6	GE 8×8R	**0.166**	0.275	0.430
7	GE-10 8×8	0.168	0.280	0.437
8	GE-11 9×9	0.167	**0.273**	**0.422**
9	AC† 10×10 SS	0.152	0.222	0.309
10	Exxon 10×10 SS	0.151	0.221	0.308
11	Damaged Dresden 1 8×8 in a DFC§	0.107	0.169	0.254
12	Dresden-1 Thin Clad 6x6§	0.124	0.212	0.343
13	Humboldt Bay-7x7§	0.127	0.215	0.343
14	Damaged Dresden-1 8x8 (in TND-1 canister) §	0.107	0.168	0.252
15	8x8 Quad+ Westinghouse§	0.164	0.278	0.435

Note: Boldface values denote the lowest thermal conductivity in each column (excluding Dresden and LaCrosse clad fuel assemblies).

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† Allis-Chalmers stainless steel clad fuel assemblies.

§ Low heat emitting fuel assemblies excluded from list of fuel assemblies (zircaloy clad) evaluated to determine the most resistive SNF type.

Table 3.4.6

MPC BASKET EFFECTIVE THERMAL CONDUCTIVITY RESULTS FROM ANSYS MODELS

Basket	@200°F (Btu/ft-hr-°F)	@450°F (Btu/ft-hr-°F)	@700°F (Btu/ft-hr-°F)
MPC-24 (Zircaloy Clad Fuel)	1.127	1.535	2.026
MPC-68 (Zircaloy Clad Fuel)	1.025	1.257	1.500
MPC-24 (Stainless Steel Clad Fuel) (Note 1)	0.901	1.230	1.615
MPC-68 (Stainless Steel Clad Fuel) (Note 1)	0.987	1.180	1.360
MPC-68 (Dresden-1 8x8 in canisters)	0.921	1.118	1.306
MPC-32 (Zircaloy Clad Fuel)	0.964	1.214	1.486
MPC-32 (Stainless Steel Clad Fuel) (Note 1)	0.762	0.936	1.104
MPC-24E (Zircaloy Clad Fuel)	1.211	1.635	2.137
MPC-24E (Stainless Steel Clad Fuel) (Note 1)	0.988	1.348	1.766

Note-1: Evaluated for a conservatively bounding configuration (fuel in a damaged fuel canister)

Table 3.4.7

INSOLATION DATA SPECIFIED BY 10CFR71, SUBPART F

Surface Type	12-Hour Total Insolation Basis	
	($g\text{-}cal/cm^2$)	($Watts/m^2$)
Horizontally Transported Flat Surfaces		
- Base	None	None
- Other Surfaces	800	774.0
Non-Horizontal Flat Surfaces	200	193.5
Curved Surfaces	400	387.0

Table 3.4.8

EFFECTIVE THERMAL CONDUCTIVITY OF THE NEUTRON SHIELD/RADIAL CHANNELS REGION

Condition/Temperature (°F)	**Thermal Conductivity (Btu/ft-hr-°F)**
Normal Condition:	
200	1.953
450	1.812
700	1.645
Fire Condition:	
200	3.012
450	2.865
700	2.689

Table 3.4.9

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Table 3.4.10

HI-STAR SYSTEM NORMAL TRANSPORT† MAXIMUM TEMPERATURES
(PWR MPCs)

	Bounding Temperature [°F]	Normal Condition Temperature Limit [°F]
Fuel Cladding	701	752
MPC Basket Centerline	667	725
MPC Basket Periphery	430	725
MPC Outer Shell Surface	315	450
MPC/Overpack Helium Gap Outer Surface	291	400
Radial Neutron Shield Inner Surface	271	300
Overpack Enclosure Shell Surface	222	350
Axial Neutron Shield	292	300
Impact Limiter Exposed Surface	121	176
Overpack Closure Plate††	163	400
Overpack Bottom Plate††	295	350

† Steady-state hot (100°F ambient) with maximum decay heat and insolation.

†† Overpack closure plate and vent/drain port plug seals normal condition design temperature is 400°F. The maximum seals temperatures are bounded by the reported closure plate and bottom plate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

Table 3.4.11

HI-STAR SYSTEM NORMAL TRANSPORT† MAXIMUM TEMPERATURES
(MPC-68)

	Bounding Temperature [°F]	**Normal Condition Temperature Limit [°F]**
Fuel Cladding	713	752
MPC Basket Centerline	697	725
MPC Basket Periphery	365	725
MPC Outer Shell Surface	306	450
MPC/Overpack Gap Outer Surface	282	400
Radial Neutron Shield Inner Surface	264	300
Overpack Enclosure Shell Surface	217	350
Axial Neutron Shield	255	300
Impact Limiter Exposed Surface	121	176
Overpack Closure Plate††	162	400
Overpack Bottom Plate††	256	350

† Steady-state hot (100°F ambient) with maximum decay heat and insolation.

†† Overpack closure plate and vent/drain port plug seals normal condition design temperature is 400°F. The maximum seals temperatures are bounded by the reported closure plate and bottom plate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

Table 3.4.12

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Table 3.4.13

SUMMARY OF BOUNDING MINIMUM
FREE VOLUME CALCULATIONS (PWR MPCs)

Item	**MPC-24 Volume (ft^3)**	MPC-24E Volume (ft^3)	MPC-32 Volume (ft^3)
Cavity Volume	367	367	367
Basket Metal Volume	45	52	25
Bounding Fuel Assemblies Volume	79	79	106
Basket Supports and Fuel Spacers Volume	7	7	9
Aluminum Conduction Elements†	6	6	6
Net Free Volume	230 (6512 liters)	223 (6314 liters)	221 (6258 liters)

† Bounding 1,000 lbs aluminum weight.

Table 3.4.14

SUMMARY OF BOUNDING MINIMUM
MPC-68 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	35
Bounding Fuel Assemblies Volume	93
Basket Supports and Fuel Spacers Volume	12
Aluminum Conduction Elements†	6
Net Free Volume	221 (6258 liters)

† Bounding 1,000 lbs aluminum weight.

Table 3.4.15

SUMMARY OF MAXIMUM NORMAL OPERATING PRESSURE (MNOP)†
FOR HORIZONTAL TRANSPORT CONDITIONS

Condition	Pressure (psig)	Bounding MPC Cavity Bulk Temperature (°F)
MPC-24:		
Initial Backfill (at 70°F)	42.8	483
Normal Condition	87.7	
With 3% Rods Rupture(Note 1)	88.8	
MPC-68:		
Initial Backfill (at 70°F)	42.8	468
Normal Condition	86.0	
With 3% Rods Rupture(Note 1)	86.9	
MPC-24E:		
Initial Backfill (at 70°F)	42.8	483
Normal Condition	87.7	
With 3% Rods Rupture(Note 1)	88.9	
MPC-32:		
Initial Backfill (at 70°F)	42.8	483
Normal Condition	87.7	
With 3% Rods Rupture(Note 1)	89.3	

Note 1: NUREG-1617 requires an assumption for normal transport that 3% of the rods are breached with release of 100% fill gas and 30% fission gas to containment.Table 3.4.16

THIS TABLE IS INTENTIONALLY DELETED.

† Pressure analysis in accordance with heat condition specified in 10 CFR 71.71(c)(1) in the absence of venting, external ancillary cooling or operational controls.

Table 3.4.17

PWR MPCs NORMAL HORIZONTAL TRANSPORT CONDITION
HI-STAR SYSTEM COMPONENTS BOUNDING TEMPERATURE [°F] SUMMARY

	MPC Basket Axial Mid-Length	MPC Basket Axial Ends
Overpack enclosure shell	222	147
Overpack inner shell	291	163
MPC shell	315	164
Basket periphery	430	166
Basket center	667	177

Table 3.4.18

MPC-68 NORMAL HORIZONTAL TRANSPORT CONDITION
HI-STAR SYSTEM COMPONENTS TEMPERATURE [°F] SUMMARY

	MPC Basket Axial Mid-Length	**MPC Basket Axial Ends**
Overpack enclosure shell	217	146
Overpack inner shell	282	161
MPC shell	306	163
Basket periphery	365	164
Basket center	697	175

Table 3.4.19

SUMMARY OF LOADED HI-STAR SYSTEM
BOUNDING COMPONENT WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
Holtite-A	11,000	0.39	4,290
Carbon Steel	140,000	0.1	14,000
Alloy-X MPC (empty)	35,000	0.12	4,200
Fuel	40,000	0.056	2,240
MPC Cavity Water†	6,500	1.0	6,500
			31,230 (Total)

† Based on smallest MPC-68 cavity net free volume with 50% credit for flooded water mass.

Table 3.4.20

MAXIMUM ALLOWABLE TIME DURATION
FOR WET TRANSFER OPERATIONS

Initial Temperature (°F)	Time Duration (hr)
115	44.3
120	42.0
125	39.7
130	37.4
135	35.2
140	32.9
145	30.6
150	28.3

Table 3.4.21

THIS TABLE IS INTENTIONALLY DELETED.

Table 3.4.22

HI-STAR SYSTEM BOUNDING TEMPERATURES [°F]
UNDER STEADY-STATE COLD† CONDITIONS (PWR MPCs)

Fuel Cladding	620
MPC Basket Centerline	586
MPC Basket Periphery	329
MPC Outer Shell Surface	190
MPC/Overpack Gap Outer Surface	165
Radial Neutron Shield Inner Surface	141
Overpack Enclosure Shell Surface	96
Axial Neutron Shield	165
Impact Limiter Exposed Surface	-40

† -40°F ambient temperature with maximum decay heat and no insolation.

Table 3.4.23

HI-STAR SYSTEM MAXIMUM TEMPERATURES [°F]
UNDER STEADY-STATE COLD† CONDITIONS (MPC-68)

Fuel Cladding	621
MPC Basket Centerline	605
MPC Basket Periphery	254
MPC Outer Shell Surface	178
MPC/Overpack Gap Outer Surface	153
Radial Neutron Shield Inner Surface	130
Overpack Enclosure Shell Surface	88
Axial Neutron Shield	123
Impact Limiter Exposed Surface	-40

† -40°F ambient temperature with maximum decay heat and no insolation.

Table 3.4.24

SUMMARY OF MPC HELIUM RETENTION BOUNDARY BOUNDING TEMPERATUREDISTRIBUTION DURING NORMAL STORAGE CONDITIONS

Location	**Figure 2.6.20 Designation**	PWR MPCs [°F]	**MPC-68 [°F]**
MPC Lid Inside Surface at Centerline	A	176	173
MPC Lid Outside Surface at Centerline	B	171	169
MPC Lid Inside Surface at Periphery	C	164	163
MPC Lid Outside Surface at Periphery	D	162	161
MPC Baseplate Inside Surface at Centerline	E	301	260
MPC Baseplate Outside Surface at Centerline	F	295	256
MPC Baseplate Inside Surface at Periphery	G	267	239
MPC Baseplate Outside Surface at Periphery	H	267	239
MPC Shell Maximum	I	315	306

Table 3.4.25

SUMMARY OF 10×10 ARRAY BWR FUEL ASSEMBLY TYPES EFFECTIVE THERMAL CONDUCTIVITIES†

Fuel	**k_{eff} at 200°F [Btu/(ft-hr-°F)]**	**k_{eff} at 450°F [Btu/(ft-hr-°F)]**	**k_{eff} at 700°F [Btu/(ft-hr-°F)]**
GE-12/14	0.166	0.269	0.412
Atrium-10	0.164	0.266	0.409
SVEA-96	0.164	0.269	0.416

† The conductivities reported in this table are obtained by the simplified method described in the beginning of Subsection 3.4.1.1.2.

Table 3.4.26

COMPARISON OF ATRIUM-10† AND BOUNDING†† BWR FUEL ASSEMBLY
EFFECTIVE THERMAL CONDUCTIVITIES

Temperature	Atrium-10 Assembly		Bounding BWR Assembly	
°F	Btu/(ft-hr-°F)	W/m-K	Btu/(ft-hr-°F)	W/m-K
200	0.225	0.389	0.171	0.296
450	0.345	0.597	0.271	0.469
700	0.504	0.872	0.410	0.710

† The reported effective thermal conductivity has been obtained from a rigorous finite-element modeling of the Atrium-10 assembly.

†† The bounding BWR fuel assembly effective thermal conductivity applied in the MPC-68 basket thermal analysis.

Table 3.4.27

THIS TABLE IS INTENTIONALLY DELETED.

Table 3.4.28

PWR MPCs BOUNDING PEAK FUEL CLADDING TEMPERATURE
AS A FUNCTION OF TOTAL HEAT LOAD

Total MPC Decay Heat Load (kW)	Peak Fuel Cladding Temperature (°F)
20.0†	700.6
19.0	678.9
17.0	633.9
15.5	598.8

† Design Basis Maximum.

Table 3.4.29

MPC-68 PEAK FUEL CLADDING TEMPERATURE
AS A FUNCTION OF TOTAL HEAT LOAD

Total MPC Decay Heat Load (kW)	Peak Fuel Cladding Temperature (°F)
18.5†	712.7
17.0	674.0
15.5	634.1

† Design Basis Maximum.

Table 3.4.30

SUMMARY OF THERMAL CONDUCTIVITY CALCULATIONS
FOR MPC HELIUM DILUTED BY RELEASED ROD GASES

Component Gas	Molecular Weight (g/mole)	Mole Fraction	Thermal Conductivity* (Btu/hr-ft-°F)
MPC and Fuel Rod Backfill Helium	4	0.817	0.098 @ 200°F 0.129 @ 450 °F 0.158 @ 700°F
Rod Tritium	3	8.007×10^{-5}	0.119 @ 200 0.148 @ 450°F 0.177 @ 700°F
Rod Krypton	85	0.016	6.76×10^{-3} @ 200°F 8.782×10^{-3} @ 450°F 0.011 @ 700°F
Rod Xenon	131	0.160	3.987×10^{-3} @ 200°F 5.258×10^{-3} @ 450°F 6.471×10^{-3} @ 700°F
Rod Iodine	129	6.846×10^{-3}	2.496×10^{-3} @ 200°F 3.351×10^{-3} @ 450°F 4.201×10^{-3} @ 700°F
Mixture of Gases (diluted helium)	N/A	1.000	0.053 @ 200°F 0.069 @ 450°F 0.085 @ 700°F

* References [3.2.2], [3.4.18] & [3.4.19] consulted for fission gases (Tritium, Krypton, Xenon and Iodine) conductivities.

Table 3.4.31

COMPARISON OF COMPONENT EFFECTIVE THERMAL CONDUCTIVITIES
WITH AND WITHOUT MPC HELIUM DILUTION

	Effective Thermal Conductivity (Btu/hr-ft-°F)		
	Value at 200°F	**Value at 450°F**	**Value at 700°F**
Fuel Assembly with Undiluted Helium	0.257	0.406	0.604
Fuel Assembly with Diluted Helium	0.160	0.278	0.458
MPC Fuel Basket with Undiluted Helium	1.127	1.535	2.026
MPC Fuel Basket with Diluted Helium	0.948	1.338	1.829

Table 3.4.32

MPC-24 HYPOTHETICAL 100% RODS RUPTURE ACCIDENT
MAXIMUM TEMPERATURES*

	Calculated Maximum Temperature (°F)	**Accident Condition Temperature Limit (°F)**
Fuel Cladding	743	1058
MPC Basket Centerline	709	950
MPC Basket Periphery	444	950
MPC Outer Shell Surface	314	775
MPC/Overpack Helium Gap Outer Surface	291	500
Radial Neutron Shield Inner Surface	271	N/A
Overpack Enclosure Shell Surface	222	1350
Overpack Closure Plate	176	700
Overpack Bottom Plate	296	700

* The results reported herein are obtained from thermal models employing grossly understated fuel basket conductivities.

Table 3.4.33

MPC-24 HYPOTHETICAL 100% RODS RUPTURE ACCIDENT PRESSURES

Calculated Accident Pressure (psig)	Accident Condition Design Pressure (psig)
134	200

Table 3.4.34

PLANT SPECIFIC BWR FUEL TYPES EFFECTIVE THERMAL CONDUCTIVITY*

Fuel	@200°F [Btu/ft-hr-°F]	@450°F [Btu/ft-hr-°F]	@700F° [Btu/ft-hr-°F]
Oyster Creek (7x7)	0.165	0.273	0.427
Oyster Creek (8x8)	0.162	0.266	0.413
TVA Browns Ferry (8x8)	0.160	0.264	0.411
SPC-5 (9x9)	0.149	0.245	0.380

* The conductivities reported in this table are obtained by a simplified analytical method described in Subsection 3.4.1.1.2.

3.6 REGULATORY COMPLIANCE

Section 3.1 defines the requirements of 10CFR71 *and ISG-11, Rev. 3 [3.1.5]) that* ~~defines the requirements and acceptance criteria, that~~ must be ~~fulfilled~~ *met* by the *HI-STAR* cask thermal design. ~~, which~~ *The cask thermal evaluations in support of these requirements* are *provided* ~~addressed~~ in Sections 3.1 through 3.5. ~~These requirements and acceptance criteria, listed in Section 3.1,~~ *In this Section, a summary of the requirements and* ~~and the conclusion~~ *results* of the evaluations are ~~summarized~~ *provided* ~~below~~.

1. The applicant must include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. The description must include, with respect to the packaging: specific materials of construction, weights, dimensions, and fabrication methods of materials specifically used as non-fissile neutron absorbers or moderators; and structural and mechanical means for the transfer and dissipation of heat. The description must include, with respect to the contents of the package: chemical and physical form; maximum normal operating pressure; maximum amount of decay heat; and identification and volumes of any coolants.

   A general description of the HI-STAR System is included in Chapter 1. Descriptions of cask materials are presented in Subsection 1.2.1, Section 1.4 and Appendices 1.A, 1.B and 1.C. Shielding materials are specifically addressed in Subsection 1.2.1.4. Cask component weights are presented in Subsections 1.2.1.1 and 2.2. Cask component dimensions are presented in Subsection 1.2.1.2 and in engineering drawings included in Section 1.4. The transfer and dissipation of heat are discussed generally in Subsection 1.2.1.6, and in detail in this chapter.

   General descriptions of and requirements for fuel assemblies for transport are presented in Subsection 1.2.3, including design basis maximum decay heat load specifications in Subsection 1.2.3.5. Maximum normal operating pressures are reported in Subsection 3.4.4. As stated in Subsection 1.2.1.7, there are no coolant volumes (reservoirs) in the HI-STAR System.

2. A package must be designed, constructed, and prepared for shipment so that under normal conditions of transport there would be no substantial reduction in the effectiveness of the packaging.

   The results of thermal evaluations presented in Section 3.4 demonstrate that the HI-STAR System performs as designed under all normal conditions of transport.

3. A package must be designed, constructed, and prepared for shipment so that in still air at 100°F and in the shade, no accessible surface of the package would have a temperature exceeding 185°F in an exclusive use shipment.

   Maximum exposed surface temperatures for the HI-STAR System are reported in Subsection 3.4.2. All impact limiter surface temperatures are shown to be below 185°F. The personnel barrier, described in Chapter 7, renders the hot overpack enclosure shell surfaces inaccessible.

4. Compliance with the permitted activity release limits for a Type B package may not depend on filters or on a mechanical cooling system.

   As stated in Section 3.1, all cooling mechanisms in the HI-STAR System are completely passive.

5. With respect to the initial conditions for the events of normal conditions of transport and hypothetical accident conditions, the demonstration of compliance with the requirements of 10CFR71 must be based on the ambient temperature preceding and following the event remaining constant at that value between -20°F and 100°F which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure (MNOP), unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the event is more unfavorable.

   Hypothetical fire accident transient calculations for the HI-STAR System are described in Section 3.5. The initial condition for this event corresponds to the most severe steady-state solution for normal conditions of transport, which correspond to a 100°F ambient temperature with full insolation. These same environmental conditions are applied during the post-accident phase of the evaluation as well. All calculated temperatures for this event are below the specified design temperature limits.

   Maximum calculated normal condition internal pressures (MNOPs) are reported in Subsection 3.4.4. Maximum calculated hypothetical accident condition internal pressures are reported in Subsection 3.5.4. All calculated MNOPs are below the design pressure limits for the MPC helium retention boundary and the overpack containment boundary.

6. For normal conditions of transport, a heat event consisting of an ambient temperature of 100°F in still air and prescribed insolation must be evaluated.

   The maximum temperatures in the HI-STAR System reported in Subsection 3.4.2 correspond to the heat event. All calculated temperatures for this event are below the appropriate design temperature limits. As stated in Subsection 3.4.5, thermal stresses are determined and reported in Chapter 2.

7. For normal conditions of transport, a cold event consisting of an ambient temperature of –40°F in still air and shade must be evaluated.

The minimum temperatures in the HI-STAR System reported in Subsection 3.4.3 correspond to the cold event. All calculated temperatures for this event are below the appropriate design temperature limits. As stated in Subsection 3.4.5, thermal stresses are determined and reported in Chapter 2.

8. Evaluation for hypothetical accident conditions is to be based on sequential application of the specified events, in the prescribed order, to determine their cumulative effect on a package.

   As described in Section 3.5, the HI-STAR System hypothetical accident thermal condition (hydrocarbon fuel/air fire) evaluation incorporates bounding representations of the results of the preceding accident conditions. Specifically, the impact limiters are assumed to be completely crushed (drop event) and the heat transfer effectiveness of the radial channels region is reduced (puncture event). All calculated temperatures for this event are below the appropriate design temperature limits.

9. For hypothetical accident conditions, a thermal event consisting of a fully engulfing hydrocarbon fuel/air fire with an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 1475°F for a period of 30 minutes.

   The description of the HI-STAR System hypothetical accident thermal event model (Subsection 3.5.1.1) specifies the fire condition input parameters. All input parameters are in accordance with the requirements of 10CFR71.73(c)(4). All calculated temperatures for this event are below the appropriate design temperature limits.

The thermal evaluations in Sections 3.4 and 3.5 demonstrate compliance with the ISG-11, Rev. 3 [3.1.5] temperature limits. Specifically, the maximum cladding temperatures for normal transport and accident conditions are below the prescribed limits (normal (752°F) and accident (1058°F)). The thermal evaluations provided in this SAR demonstrate that the HI-STAR System description and evaluation satisfy the thermal requirements of 10 CFR Part 71. Specifically:

- The material properties and component specifications used in the thermal evaluation are sufficient to provide a basis for evaluation of the HI-STAR System against the thermal requirements of 10 CFR Part 71.

- The methods used in the thermal evaluation are described in sufficient detail to permit an independent review, with confirmatory calculations, of the HI-STAR System thermal design.

- The accessible surface temperatures of the HI-STAR System as it will be prepared for shipment satisfy 10 CFR 71.43(g) for exclusive use shipments.

- The HI-STAR System design, construction, and preparations for shipment ensure that the material and component temperatures will not extend beyond the specified allowable limits during normal conditions of transport consistent with 10 CFR 71.71.

- The HI-STAR System design, construction, and preparations for shipment ensure that the material and component temperatures will not exceed the specified allowable temperature limits during hypothetical accident conditions consistent with 10 CFR 71.73.

It is therefore concluded that the thermal design of the HI-STAR System is in compliance with 10 CFR Part 71, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the HI-STAR System will allow safe transport of spent fuel. This conclusion is based on the technical data and analyses presented in this chapter in conjunction with provisions of 10 CFR Part 71, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

3.7 REFERENCES

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4.2 REQUIREMENTS FOR NORMAL AND HYPOTHETICAL ACCIDENT CONDITIONS OF TRANSPORT

Chapter 2 shows that all primary and secondary containment components are maintained within their code-allowable stress limits during all normal and hypothetical accident conditions of transport as defined in 10CFR71.71 and 10CFR71.73 [4.0.1]. Chapter 3 shows that the peak containment component temperatures and pressure are within the design basis limits for all normal and hypothetical accident conditions of transport as defined in 10CFR71.71 and 10CFR71.73. Since the primary and secondary containment vessels remain intact, and the temperature and pressure design bases are not exceeded, the design basis leakage rate (see Table 4.1.1) will not be exceeded during normal or hypothetical accident conditions of transport.

4.2.1 Containment Criteria

The allowable leakage rates presented in this chapter were determined in accordance with ANSI N14.5-1997 [4.0.2] and shall be used for containment system fabrication verification and containment system periodic verification tests of the HI-STAR 100 containment boundaries. Measured leakage rates shall not exceed the values presented in Table 4.1.1. Compliance with these leakage rates ensures that the radionuclide release rates specified in 10CFR71.51 and 10CFR71.63(b) will not be exceeded during normal or hypothetical accident conditions of transport.

4.2.2 Containment of Radioactive Material

The HI-STAR 100 packaging allowable leakage rate (See Table 4.1.1) ensures that the requirements of 10CFR71.51 and 10CFR71.63(b) are met. Section 4.2.5 determines the maximum leakage rate for normal and hypothetical accident conditions of transport and the allowable leakage rate criterion for the HI-STAR 100 packaging containing each of the MPC types. The maximum calculated leakage rates for normal transport conditions assume a full complement of design basis fuel assembly types with bounding radiological source terms. The calculations also assume 3% fuel rod rupture for normal conditions. This bounds all possible MPC fuel loading configurations. For calculating the maximum leakage rates for normal conditions of transport, the internal pressure is conservatively assumed to be greater than the MPC internal pressure for the most limiting MPC type determined in Chapter 3. Following testing, no credit is taken for the MPC as a containment boundary for the transport of intact fuel. The MPC enclosure vessel is identified as the secondary containment boundary for the transport of the specified fuel debris in accordance with the 10CFR71.63(b) requirements for a separate inner container.

The allowable leakage rate is then conservatively chosen to be less than the calculated maximum leakage rates from all MPC types for normal conditions of transport. This ensures that the 10CFR71.51(a)(1) and 71.63(b) limits for radionuclide release are not exceeded.

4.2.3 Pressurization of Containment Vessel

The HI-STAR 100 overpack contains a sealed MPC during normal conditions of transport. Except for the small space between the MPC and overpack, the overpack internal cavity is essentially filled.

This space (annulus) is drained, dried, evacuated and backfilled with helium gas prior to final closure of the overpack; therefore, no vapors or gases are present which could cause a reaction or explosion inside the overpack. Procedural steps (Chapter 7) prevent overpack over-pressurization during closure operations. The enclosed MPC is also drained, dried, and backfilled with helium gas prior to final closure; therefore, any MPC leak would not introduce any explosive gases into the overpack cavity. Since the exterior of the MPC is entirely composed of stainless steel, there is no possibility of chemical reaction that would produce gas or vapor. The overpack accident condition design basis internal pressure analysis assumes a non-mechanistic event resulting in the loss of MPC closure welds, a full-complement of design basis fuel with 100% fill gas and 30% of significant fission gas release, and the hypothetical 10CFR71.73(c)(4) fire condition. Even in this event, structural integrity and containment of the HI-STAR 100 packaging are maintained.

As the MPC is drained, dried, evacuated and backfilled with helium gas, no vapors or gases are present which could cause a reaction or explosion inside the MPC. Procedural steps (Chapter 7) prevent MPC over-pressurization during closure operations. The interior of the MPC contains stainless steel, Boral, and optional aluminum heat conductive inserts. There is no possibility of chemical reaction that would produce gas or vapor.

4.2.4 Assumptions

The HI-STAR 100 System is designed to meet the radioactive release limit requirements of 10CFR71.51 and 10CFR71.63(b). Allowable leakage rates are determined in accordance with the requirements of ANSI N14.5, and utilizing NUREG/CR-6487, *Containment Analysis for Type B Packages Used to Transport Various Contents* [4.0.3] and Regulatory Guide 7.4, *Leakage Tests on Packages for Shipment of Radioactive Materials* [4.0.4] as guides.

The following assumptions have been used in determining the allowable leakage rates:

1. For MPCs other than the MPC-24EF with Trojan fuel debris and MPC-68F, three percent of the fuel rods are assumed to have failed during normal conditions of transportation. One-hundred percent of the fuel rods are assumed to have failed during hypothetical accident conditions.

2. Thirty percent of the radioactive gases are assumed to escape each failed fuel rod.

3. Fifteen percent of the ^{60}Co from the crud on the surface of the fuel rods is released as an aerosol in normal conditions of transport. One-hundred percent of the ^{60}Co is released as an aerosol from the surfaces of the fuel assemblies during accident conditions.

4. Since the overpack internals are never exposed to contaminants, the residual activity on the overpack interior surface and the MPC exterior surface is negligible compared to crud deposits on the fuel and is neglected as a source term.

5. Up to four (4) DFCs containing specified fuel debris may be placed in an MPC-24EF (only the custom-designed Trojan MPC-24EF) or an MPC-68F.

6. Crud spallation and cladding breaches occur instantaneously after fuel loading and container closure operations.

7. The calculation for normal transport conditions of an MPC containing fuel debris assumes 100% of the rods of the fuel debris are breached.

8. For containment analysis purposes, the MPC-24, MPC-24E or MPC-24EF contain up to 24 PWR assemblies, of which 4 of these in the custom-designed Trojan MPC-24EF may be DFCs with Trojan fuel debris, the MPC-32 contains up to 32 PWR assemblies, the MPC-68 contains up to 68 BWR assemblies, and the MPC-68F contains up to 68 intact BWR fuel assemblies, of which 4 of those may be specified BWR fuel debris in damaged fuel containers.

9. 0.003% of the total fuel mass contained in a rod is assumed to be released as fines if the cladding on the rod ruptures (i.e., $f_f=3x10^{-5}$).

10. Bounding values for the crud surface activity for PWR rods is $140x10^{-6}$ Ci/cm^2 and for BWR rods is $1254x10^{-6}$ Ci/cm^2.

11. The rod surface area per assembly is $3x10^5$ cm^2 for PWR and $1x10^5$ cm^2 for BWR fuel assemblies. These surface areas are also conservatively used for the surface area of damaged fuel or fuel debris..

12. The release fractions for volatiles (^{89}Sr, ^{90}Sr, ^{103}Ru, ^{106}Ru, ^{134}Cs, ^{135}Cs, and ^{137}Cs) are all assumed to be $2x10^{-4}$ ($f_V=2x10^{-4}$).

13. In the analysis of the primary containment boundary, the MPC is assumed to rupture. In the analysis of the secondary containment boundary, the primary containment is assumed to fail.

14. In calculating the leakage rates of the primary containment for normal conditions of transport, the internal pressure of the overpack is conservatively assumed to be larger than *or equal to* the maximum internal pressure of all MPC types determined in Chapter 3.

15. The average cavity temperature for ~~all analyses~~ *accident conditions* is conservatively assumed to be the design basis peak cladding temperature *of 1058°F (843K)*.

16. All of the activity associated with crud is assumed to be Cobalt-60.

17. It is assumed that the flow is unchoked for all leakage analyses.

18. In the evaluation to demonstrate compliance with 10CFR71.63(b), the source activity due to Plutonium was determined by conservatively assuming that all of the rods develop cladding breaches during normal transportation and hypothetical accident conditions (i.e., $f_B=1.0$).

19. In the evaluation to demonstrate compliance with 10CFR71.63(b), the assumption was also made that roughly 0.003% of the plutonium is released from a fuel rod (i.e., $f_{Pu}=3x10^{-5}$).

4.2.5 Analysis and Results

The allowable leakage rates for the primary and secondary containment boundaries under normal and hypothetical accident conditions of transport at operating conditions for the HI-STAR 100 packaging containing each of the MPC types were determined and are presented in this chapter. To calculate the leakage rates for a particular contents type and transportation condition, the following were determined: the source term concentration for the releasable material; the effective A_2 of the individual contributors; the releasable activity; the effective A_2 for the total source term; the allowable radionuclide release rates; and the allowable leakage rates at transport (operating) conditions. Using the equations for continuum and molecular flow, the corresponding leakage hole diameters were calculated. Then, using these leak hole diameters, the corresponding allowable leakage rates at test conditions were calculated. Parameters were utilized in a way that ensured conservatism in the final leakage rates for the conditions, contents, and package arrangements considered.

The methodology and analysis results are summarized below.

4.2.5.1 Volume in the Containment Vessel

As discussed above, the primary containment system boundary for the HI-STAR 100 packaging consists of the overpack inner shell and associated components and the secondary containment system boundary consists of the MPC enclosure vessel and associated components. The MPC provides the separate inner container per 10CFR71.63(b) for the HI-STAR 100 System transporting fuel classified as fuel debris.

Except for a small volume between the MPC and the overpack (the annulus), the overpack internal cavity is essentially filled. Therefore, the free gas volume for the primary containment boundary includes the free gas volume for the MPC plus the overpack annulus volume. The free gas volume in each of the MPC types is presented in Chapter 3. The free gas volumes of the primary and secondary containment are repeated in Table 4.2.1 for completeness. The MPC-24E and MPC-24EF basket designed for Trojan are shorter to allow for storage in their overpacks. These shorter baskets are designated as the Trojan MPC-24E and Trojan MPC-24EF, respectively, where necessary. For calculating the free volume in the primary containment (overpack) with either of the Trojan MPCs, the annulus space is assumed to be the same as that for the larger generic MPCs (i.e. the larger annulus space between the Trojan MPC and HI-STAR overpack is neglected). This will conservatively underestimate the free volume inside the primary containment.

4.2.5.2 Source Terms For Spent Nuclear Fuel Assemblies

In accordance with NUREG/CR-6487 [4.0.3], the following contributions are considered in determining the releasable source term for packages designed to transport irradiated fuel rods: (1)

the radionuclides comprising the fuel rods, (2) the radionuclides on the surface of the fuel rods, and (3) the residual contamination on the inside surfaces of the vessel. NUREG/CR-6487 goes on to state that a radioactive aerosol can be generated inside a vessel when radioactive material from the fuel rods or from the inside surfaces of the container become airborne. The sources for the airborne material are (1) residual activity on the cask interior, (2) fission and activation-product activity associated with corrosion-deposited material (crud) on the fuel assembly surface, and (3) the radionuclides within the individual fuel rods. In accordance with NUREG/CR-6487, contamination due to residual activity on the cask interior surfaces is negligible as compared to crud deposits on the fuel rods themselves and therefore may be neglected. The source term considered for this calculation results from the spallation of crud from the fuel rods and from the fines, gases and volatiles which result from cladding breaches.

The inventory for isotopes other than ^{60}Co is calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system as described in Chapter 5. The inventory for the MPC-24, MPC-24E, MPC-24EF, and MPC-32 was conservatively based on the B&W 15x15 fuel assembly with a burnup of 45,000 MWD/MTU, 5 years of cooling time, and an enrichment of 3.6%. The inventory for the Trojan MPCs (Trojan MPC-24E, Trojan MPC-24EF) was based on the Westinghouse 17x17 fuel assembly with a burnup of 42,000 MWD/MTU, 9 years cooling time, and an enrichment of 3.09%. The inventory for the MPC-68 was based the GE 7x7 fuel assembly with a burnup of 45,000 MWD/MTU, 5 years of cooling time, and 3.2% enrichment. The inventory for the MPC-68F was based on the GE 6x6 fuel assembly with a burnup of 30,000 MWD/MTU, 18 years of cooling time, and 1.8% enrichment. Additionally, an MPC-68F was analyzed containing 67 GE 6x6 assemblies and a DFC containing 18 thorium rods. Finally, an Sb-Be source stored in one fuel rod in one assembly with 67 GE 6x6 assemblies was analyzed. The isotopes which contribute greater than 0.01% to the total curie inventory for the fuel assembly are considered in the evaluation as fines. Additionally, isotopes with A_2 values less than 1.0 in Table A-1, Appendix A, 10CFR71 are included as fines. Isotopes which contribute greater than 0.01% but which do not have an assigned A_2 value in Table A-1 are assigned an A_2 value based on the guidance in Table A-2, Appendix A, 10CFR71. ~~Isotopes which contribute greater than 0.01% but have a radiological half life less than 10 days are neglected.~~*Finally, those radionuclides that have no A_2 value in Table A-1 from Appendix A of 10CFR71, have a half-life shorter than 10 days, and have a half-life less than their parent radionuclide (i.e., are in secular equilibrium with their parent nuclide), are in accordance with 10CFR71, Appendix A, III treated as a single radionuclide along with the parent nuclide.* Table 4.2.2 presents the isotope inventory used in the calculation.

A. <u>Source Activity Due to Crud Spallation from Fuel Rods</u>

The majority of the activity associated with crud is due to ^{60}Co [4.0.3]. The inventory for ^{60}Co was determined by using the crud surface activity for PWR rods (140×10^{-6} Ci/cm^2) and for BWR rods (1254×10^{-6} Ci/cm^2) provided in NUREG/CR-6487 [4.0.3] multiplied by the surface area per assembly (3×10^5 cm^2 and 1×10^5 cm^2 for PWR and BWR, respectively, also provided in NUREG/CR-6487).

The source terms were then decay corrected (5 years for the MPC-24, MPC-24E, MPC-24EF, MPC-32 and the MPC-68; 18 years for the MPC-68F; 9 years for the Trojan MPCs) using the basic radioactive decay equation:

$$A(t) = A_0 e^{-\lambda t} \tag{4-1}$$

where:

A(t)	is activity at time t [Ci]
A_o	is the initial activity [Ci]
λ	is the $\ln 2/t_{1/2}$ (where $t_{1/2}$ = 5.272 years for ^{60}Co)
t	is the time in years (5 years for the MPC-24, MPC-24E, MPC-24EF, MPC-32 and the MPC-68; 18 years for the MPC-68F; 9 years for the Trojan MPCs)

The inventory for ^{60}Co was determined using the methodology described above with the following results:

PWR
Surface area per Assy = 3.0E+05 cm^2
140 $\mu Ci/cm^2$ x 3.0E+05 cm^2 = 42.0 Ci/assy

BWR
Surface area per Assy = 1.0E+05 cm^2
1254 $\mu Ci/cm^2$ x 1.0E+05 cm^2 = 125.4 Ci/assy

$^{60}Co(t) = {}^{60}Co_0\, e^{-(\lambda t)}$, where $\lambda = \ln 2/t_{1/2}$, t = 5 years (for the MPC-24, MPC-24E, MPC-24EF, MPC-32 and MPC-68), t = 18 years (MPC-68F), t = 9 years (Trojan MPCs), $t_{1/2}$ = 5.272 years for ^{60}Co [4.2.4]

MPC-24, MPC-24E, MPC-24EF, MPC-32
$^{60}Co(5) = 42.0\ Ci\ e^{-(\ln 2/5.272)(5)}$
$^{60}Co(5)$ = 21.77 Ci/assy

MPC-68
$^{60}Co(5) = 125.4\ Ci\ e^{-(\ln 2/5.272)(5)}$
$^{60}Co(5)$ = 64.98 Ci/assy

Trojan MPC-24E, Trojan MPC-24EF
$^{60}Co(5) = 42.0\ Ci\ e^{-(\ln 2/5.272)(9)}$
$^{60}Co(5)$ = 12.86 Ci/assy

MPC-68F
$^{60}Co(18) = 125.4\ Ci\ e^{-(\ln 2/5.272)(18)}$
$^{60}Co(18)$ = 11.76 Ci/assy

A summary of the ^{60}Co inventory available for release is provided in Table 4.2.2.

The activity density that results inside the containment vessel as a result of crud spallation from spent fuel rods can be formulated as:

$$C_{crud} = \frac{f_C M_A N_A}{V} \tag{4-2}$$

where:

C_{crud}	is the activity density inside the containment vessel as a result of crud spallation [Ci/cm^3],
M_A	is the total crud activity inventory per assembly [Ci/assy],
f_C	is the crud spallation fraction,
N_A	is the number of assemblies, and
V	is the free volume inside the containment vessel [cm^3].

NUREG/CR-6487 states that measurements have shown 15% to be a reasonable value for the percent of crud spallation for both PWR and BWR fuel rods under normal transportation conditions. For hypothetical accident conditions, it is assumed that there is 100% crud spallation [4.0.3].

B. Source Activity Due to Releases of Fines from Cladding Breaches

A breach in the cladding of a fuel rod may allow radionuclides to be released from the resulting cladding defect into the interior of the MPC. If there is a leak in the primary or secondary containment vessels, then the radioisotopes emitted from a cladding breach that were aerosolized may be entrained in the gases escaping from the package and result in a radioactive release to the environment.

NUREG/CR-6487 suggests that a bounding value of 3% of the rods develop cladding breaches during normal transportation (i.e., f_B=0.03). For hypothetical accident conditions, it is assumed that all of the rods develop a cladding breach (i.e., f_B=1.0). These values were used for both PWR and BWR fuel rods. As described in NUREG/CR-6487, roughly 0.003% of the fuel mass contained in a rod is released as fines if the cladding on the rod ruptures (i.e., $f_f=3\times10^{-5}$).

The calculation for normal transport conditions of either a Trojan MPC-24EF or an MPC-68F containing four (4) DFCs containing fuel debris assumes that for the four DFCs, 100% of the rods of the fuel debris are breached. The remaining 20 or 64 assemblies in either the Trojan MPC-24EF or the MPC-68F, respectively, were assumed to have a 3% cladding rupture. Therefore, f_B for a Trojan MPC-24EF or an MPC-68F containing fuel debris is:

$$f_B = (0.03)\frac{20}{24} + (1.0)\frac{4}{20} \qquad \text{(4-3a)}$$
$$f_B = 0.192$$

$$f_B = (0.03)\frac{64}{68} + (1.0)\frac{4}{68} \qquad \text{(4-3b)}$$
$$f_B = 0.087$$

The activity concentration inside the containment vessel due to fines being released from cladding breaches is given by:

$$C_{fines} = \frac{f_f\, I_{fines}\, N_A\, f_B}{V} \qquad \text{(4-4)}$$

where:

C_{fines} is the activity concentration inside the containment vessel as a result of fines released from cladding breaches [Ci/cm^3],

f_f	is the fraction of a fuel rod's mass released as fines as a result of a cladding breach (f_f=3x10^{-5}),
I_{fines}	is the total activity inventory [Ci/assy],
N_A	is the number of assemblies,
f_B	is the fraction of rods that develop cladding breaches, and
V	is the free volume inside the containment vessel [cm^3].

C. Source Activity from Gases due to Cladding Breaches

If a cladding failure occurs in a fuel rod, a large fraction of the gap fission gases will be introduced into the free volume of the system. Tritium and Krypton-85 are typically the major sources of radioactivity among the gases present [4.0.3]. NUREG/CR-6487 suggests that a bounding value of 30% of the fission product gases escape from a fuel rod as a result of a cladding breach (i.e., f_g=0.3).

The activity concentration due to the release of gases form a cladding breach is given by:

$$C_{gases} = \frac{f_g\, I_{gases}\, N_A\, f_B}{V} \qquad (4\text{-}5)$$

where:

C_{gases}	is the releasable activity concentration inside the containment vessel due to gases released from cladding breaches [Ci/cm^3],
f_g	is the fraction of gas that would escape from a fuel rod that developed a cladding breach,
I_{gases}	is the gas activity inventory [^3H, ^{129}I, ^{85}Kr, ^{81}Kr, ^{127}Xe] [Ci/assy],
N_A	is the number of assemblies,
f_B	is the fraction of rods that develop cladding breaches, and
V	is the free volume inside the containment vessel [cm^3].

D. Source Activity from Volatiles due to Cladding Breaches

Volatiles such as cesium, strontium, and ruthenium, can also be released from a fuel rod as a result of a cladding breach. NUREG/CR-6487 estimates that 2x10^{-4} is a conservative bounding value for the fraction of the volatiles released from a fuel rod (i.e., f_V=2x10^{-4}).

The activity concentration due to the release of volatiles is given by:

$$C_{vol} = \frac{f_v\, I_{vol}\, N_A\, f_B}{V} \qquad (4\text{-}6)$$

where:

C_{vol}	is the releasable activity concentration inside the containment vessel due to volatiles released from cladding breaches [Ci/cm^3],
f_V	is the fraction of volatiles that would escape from a fuel rod that developed a cladding breach,

I_{vol} is the volatile activity inventory [^{89}Sr, ^{90}Sr, ^{134}Cs, ^{135}Cs, ^{137}Cs, ^{134}Cs, ^{103}Ru, ^{106}Ru] [Ci/assy],
N_A is the number of assemblies,
f_B is the fraction of rods that develop cladding breaches, and
V is the free volume inside the containment vessel [cm^3].

E. Total Source Term for the HI-STAR 100 System

The total source term was determined by combining Equations 4-2, 4-4, 4-5, and 4-6:

$$C_{total} = C_{crud} + C_{fines} + C_{gases} + C_{vol} \tag{4-7}$$

where C_{total} has units of Ci/cm^3.

Table 4.2.3 presents the total source term determined using the above methodology. Table 4.2.4 summarizes the parameters from NUREG/CR-6487 used in this analysis.

4.2.5.3 Effective A_2 of Individual Contributors (Crud, Fines, Gases, and Volatiles)

The A_2 of the individual contributions (i.e., crud, fines, gases, and volatiles) were determined in accordance with NUREG/CR-6487. As previously described, the majority of the activity due to crud is from Cobalt-60. Therefore, the A_2 value of 10.8 Ci used for crud for both PWR and BWR fuel is the same as that for Cobalt-60 found in 10CFR71, Appendix A.

In accordance with 10CFR71.51(b) the methodology presented in 10CFR71, Appendix A for mixtures of different radionuclides was used to determine the A_2 values for the gases, fines and volatiles.

$$A_2 \text{ for a mixture} = \frac{1}{\sum_{i=1}^{I} \frac{f_i}{(A_2)_i}} \tag{4-8}$$

Where f(i) is the fraction of activity of nuclide I in the mixture and $A_2(i)$ is the appropriate A_2 value for the nuclide I.

10CFR71.51(b) also states that for Krypton-85, an effective A_2 value equal to 10 A_2 may be used. Table 4.2.5 summarizes the effective A_2 for all individual contributors.

4.2.5.4 Releasable Activity

The releasable activity is the product of the respective activity concentrations (C_{fines}, C_{gas}, C_{crud}, and C_{vol}) and the respective MPC volume. The releasable activity of fines, volatiles, gases, and crud were determined using this methodology.

$$\text{Releasable Activity [Ci]} = \text{Activity Concentration } [\frac{\text{Ci}}{\text{cm}^3}] \text{ x Volume } [\text{cm}^3] \qquad \textbf{(4-9)}$$

4.2.5.5 Effective A_2 for the Total Source Term

Using the releasable activity and the effective A_2 values from the individual contributors (i.e., crud, fines, gases, and volatiles), the effective A_2 for the total source term was calculated for each MPC type, for normal transportation and hypothetical accident conditions. The methodology used to determine the effective A_2 is the same as that used for a mixture, which is provided in Equation 4-8.

The results are summarized in Table 4.2.6. As stated in 4.2.5.3, the effective A_2 used for Krypton-85 is 10 A_2 (2700 Ci).

4.2.5.6 Allowable Radionuclide Release Rates

The containment criterion for the HI-STAR 100 System under normal conditions of transport is given in 10CFR71.51(a)(1). This criterion requires that a package have a radioactive release rate less than $A_2 \times 10^{-6}$ in one hour, where A_2 is the effective A_2 for the total source term in the packaging determined in 4.2.5.5. Additionally, 10CFR71.51(b)(2) specifies that for hypothetical accident conditions, the quantity that may be released in one week is A_2 (effective A_2 for the total source term determined in 4.2.5.5).

NUREG/CR-6487 and ANSI N14.5 provides the following equations for the allowable release rates.

Release rate for normal conditions of transport:

$$R_N = L_N C_N \leq A_2 \times 2.78\times10^{-10}/\text{second} \qquad \textbf{(4-10)}$$

where:

R_N is the release rate for normal transport [Ci/s]
L_N is the volumetric gas leakage rate [cm^3/s]
C_N is the total source term activity concentration [Ci/cm^3]
A_2 is the appropriate effective A_2 value [Ci].

Release rate for hypothetical accident conditions:

$$R_A = L_A C_A \leq A_2 \times 1.65\times10^{-6}/\text{second} \qquad \textbf{(4-11)}$$

where:

R_A	is the release rate for hypothetical accident conditions [Ci/s]
L_A	is the volumetric gas leakage rate [cm^3/s]
C_A	is the total source term activity concentration [Ci/cm^3]
A_2	is the appropriate effective A_2 value [Ci].

Equations 4-10 and 4-11 were used to determine the allowable radionuclide release rates for each MPC type and transport condition. The release rates are summarized in Table 4.2.7.

4.2.5.7 Allowable Leakage Rates at Operating Conditions

The allowable leakage rates at operating conditions were determined by dividing the allowable release rates by the appropriate source term activity concentration (modifying Equations 4-10 and 4-11).

$$L_N = \frac{R_N}{C_N} \text{ or } L_A = \frac{R_A}{C_A} \tag{4-12}$$

where,

L_N or L_A	is the allowable leakage rate at the upstream pressure for normal (N) or accident (A) conditions [cm^3/s],
R_N or R_A	is the allowable release rate for normal (N) or accident (A) conditions [Ci/s], and
C_N or C_A	is the allowable release rate for normal (N) or accident (A) conditions [Ci/cm^3].

The allowable leakage rates determined using Equation 4-12 are the allowable leakage rates at the upstream pressure. Table 4.2.9 summarizes the allowable leakage rates at the upstream pressures. The most limiting allowable leakage rate presented in Table 4.2.9 was conservatively selected and used to determine the leakage rate acceptance criterion.

Equation deleted (4-13)

4.2.5.8 Leakage Rate Acceptance Criteria for Test Conditions

The leakage rates discussed thus far were determined at operating conditions (see normal and accident conditions in Table 4.2.12). The following provides details of the methodology used to convert the allowable leakage rate at operating conditions to a leakage rate acceptance criterion at reference test conditions.

For conservatism, unchoked flow correlations were used as the unchoked flow correlations better approximate the true measured flow rate for the leakage rates associated with transportation packages. Using the equations for molecular and continuum flow provided in NUREG/CR-6487, the corresponding leak hole diameter was calculated by solving Equation 4-14a for D, the leak hole diameter. The capillary length required for Equation 4-14a for the primary containment was conservatively chosen as the closure plate inner seal seating width which is 0.25 cm; for the secondary containment, the capillary length was conservatively chosen to be the MPC lid closure weld thickness which is 1.25 inches thick (3.175 cm).

$$L_{@P_u} = \left[\frac{2.49 \times 10^6 \, D^4}{a \, u} + \frac{3.81 \times 10^3 \, D^3 \sqrt{\frac{T}{M}}}{a \, P_a} \right] \left[P_u - P_d \right] \frac{P_a}{P_u} \qquad \textbf{(4-14a)}$$

where:

$L_{@Pu}$ is the allowable leakage rate at the upstream pressure for normal and accident conditions [cm^3/s],
a is the capillary length [cm],
T is the temperature for normal and accident conditions [K],
M is the gas molecular weight [g/mole] = 4.0 from ANSI N14.5, Table B1 [4.0.2],
u is the fluid viscosity for helium [cP] from Rosenhow and Hartnett [4.2.3]
P_u is the upstream pressure [ATM],
D leak hole diameter [cm],
P_d is the downstream pressure for normal and accident conditions [ATM], and
P_a is the average pressure; $P_a = (P_u + P_d)/2$ for normal and accident conditions [ATM].

The actual leakage tests performed on the primary and secondary confinement boundary welds are typically not performed under exactly the same conditions every time. Therefore, reference test conditions are specified to provide a consistent comparison of the measured leakage rate to the leakage rate acceptance criterion. For example, the MPC Lid-to-Shell weld is performed with an elevated pressure (85 psig min) inside the MPC cavity to magnify the leakage rate in the event of a leak. The reference test conditions, and approximate actual test conditions are specified in Table 4.2.12.

The corresponding leak hole diameter at operating conditions was determined by solving Equation 4-14a for 'D' where $L_{@Pu}$ is equal to 1.03×10^{-5} cm^3/s and using the parameters for normal conditions of transport presented in Table 4.2.12.

Using this leak hole diameter and the temperature and pressure specified for reference test conditions provided in Table 4.2.12, Equation 4-14a was solved for the volumetric leakage rate at reference test conditions.

Equation B-1 of ANSI N14.5-1997 [4.0.2] is used to express this volumetric leakage rate into a mass-like helium flow rate (Q_u) as follows:

$$Q_u = L_u * P_u \ \ (\text{atm-cm}^3/\text{sec}) \qquad \textbf{(4-14b)}$$

where:

L_u is the upstream volumetric leakage rate [cm^3/sec],
Q_u is the mass-like helium leak rate [$atm\text{-}cm^3/sec$], and
P_u is the upstream pressure [atm].

Using Equation 4-14b to convert the volumetric flow rate into a mass-like flow, the leakage rate acceptance criteria is calculated to be 5.41×10^{-6} atm-cm^3/sec, which has been conservatively reduced and is presented in Table 4.1.1.

Table 4.2.12 provides additional parameters used in the analysis.

4.2.5.9 10CFR71.63(b) Plutonium Leakage Verification

The HI-STAR 100 System configured to transport fuel debris must meet the criteria of 10CFR71.63(b) for plutonium shipments. This criteria specifies that for normal conditions of transport, the separate inner container must not release plutonium as demonstrated to a sensitivity of $A_2 \times 10^{-6}$ in one hour, where A_2 is the effective A_2 for the plutonium inventory in the damaged fuel (up to four DFCs containing specified fuel debris). Additionally, 10CFR71.63(b) specifies that for hypothetical accident conditions, the separate inner container must restrict the loss of plutonium to not more than A_2 in one week (effective A_2 for the plutonium inventory determined using the methodology described in Section 4.2.5.3).

To demonstrate compliance with this requirement, the leakage rate acceptance criterion was determined following the basic methodology described above. To determine this leakage rate, the plutonium inventory for the GE 6x6 MOX fuel assembly and the plutonium inventories for the assemblies described in Section 4.2.5.2 was analyzed. Table 4.2.11 contains the plutonium inventory for the MOX fuel used in this evaluation.

As discussed in 4.2.5.2, Equation 4-3a and Equation 4-3b presents the methodology to determine f_B for a Trojan MPC-24EF and an MPC-68F containing fuel debris, respectively. This f_B was applied in determining the source activity due to Plutonium. The calculation for normal transport conditions of an MPC containing four (4) DFCs containing fuel debris assumes that for the four DFCs, 100% of the rods of the fuel debris are breached. The remaining assemblies in the MPC were assumed to have a 3% cladding rupture. The source activity due to Plutonium was determined by conservatively assuming that all of the rods develop cladding breaches during hypothetical accident conditions (i.e., f_B=1.0). The assumption was also made that roughly 0.003% of the plutonium is released from a fuel rod (i.e., $f_{Pu}=3 \times 10^{-5}$). Therefore, the activity concentration inside the containment vessel due to plutonium is given by:

$$C_{Pu} = \frac{f_{Pu} I_{Pu} N_A f_B}{V} \qquad \textbf{(4-15)}$$

where:

C_{Pu} is the activity concentration inside the containment vessel from Plutonium [Ci/cm^3],
f_{Pu} is the fraction of a fuel rod's mass released as Plutonium ($f_f = 3x10^{-5}$),
I_{Pu} is the total Plutonium inventory of one assembly [Ci/assy],
N_A is the number of assemblies,
f_B is the fraction of rods that develop cladding breaches (f_B=0.087 for BWR fuel and f_B=0.192 for PWR fuel under normal conditions of transport and f_B=1.0 for accident conditions), and
V is the free volume inside the containment vessel [cm^3] from Table 4.2.1.

The methodology described in 4.2.5.3 for mixtures was used to calculate the effective A_2 for Plutonium. The methodology in 4.2.5.4 was used to determine the releasable activity. The allowable radionuclide release rates were determined using the methodology presented in 4.2.5.6 and are summarized in Table 4.2.13. The allowable leakage rates at the upstream pressure were determined as discussed in 4.2.5.7 (using Equation 4-12). The allowable leakage rates are presented in Table 4.2.14. As in 4.2.5.7, the most limiting allowable leakage rate presented in Table 4.2.14 was conservatively selected and used to determine the leakage rate acceptance criterion for the MPC.

As discussed in 4.2.5.8, the allowable leakage rate was then converted to a leakage rate acceptance criterion at test conditions using the equations for molecular and continuum flow provided in NUREG/CR-6487 (Equation 4-14a). The capillary length required for Equation 4-14a for the secondary containment was conservatively chosen to be the MPC lid closure weld thickness which is assumed to be 1.25 inches thick (3.175 cm). Equation 4-14a was solved for D, the leak hole diameter and then using this leak hole diameter, and the temperature and pressures for test conditions (Table 4.1.12), Equation 4-14a was solved for the volumetric leakage rate acceptance criterion at test conditions. Equation 4-14b is used to convert the volumetric flow rate into the mass-like flow rate, resulting in an acceptance criterion leakage rate of $8.94x10^{-6}$ atm-cm^3/sec. For additional conservatism to ensure compliance with 10CFR71.63(b), this leakage rate acceptance criterion was conservatively reduced and is presented in Table 4.1.1.

4.2.5.10 Leak Test Sensitivity

The sensitivity for the overpack leakage test procedures is equal to one-half of the allowable leakage rate. The HI-STAR 100 containment packaging tests in Chapter 8 incorporate the appropriate leakage test procedure sensitivity. The leakage rates for the HI-STAR 100 containment packaging with its corresponding sensitivity are presented in Table 4.1.1.

Table 4.2.1

FREE GAS VOLUME OF THE PRIMARY AND SECONDARY CONTAINMENT

MPC Type	Primary Containment Volume (overpack) (cm^3)	Secondary Containment Volume (MPC) (cm^3)
MPC-24	6.70×10^6	N/A
MPC-24E MPC-24EF	6.55×10^6	N/A
Trojan MPC-24E Trojan MPC-24EF	6.12×10^6	5.96×10^6
MPC-32	6.35×10^6	N/A
MPC-68	6.15×10^6	N/A
MPC-68F	6.15×10^6	5.99×10^6

Table 4.2.2

ISOTOPE INVENTORY
Ci/Assembly

Nuclide	PWR MPCs Ci/Assembly	MPC-68 Ci/Assembly	MPC-68F Ci/Assembly	Trojan MPCs Ci/Assembly
Gases				
^{3}H	2.76E+02	1.09E+02	1.78E+01	1.75E+02
^{129}I	2.17E-02	8.66E-03	3.49E-03	1.93E-02
^{85}Kr	4.69E+03	1.79E+03	2.37E+02	2.76E+03
^{81}Kr	7.97E-08	3.50E-08	1.19E-08	6.80E-08
^{127}Xe	5.95E-11	2.05E-11	1.62E-17	3.39E-29
Crud				
^{60}Co	2.18E+01	6.50E+01	1.18E+01	1.29E+01
Volatiles				
^{90}Sr	4.53E+04	1.76E+04	4.29E+03	3.36E+04
^{106}Ru	4.97E+04	1.74E+04	2.30E-01	7.99E+02
^{134}Cs	4.43E+04	1.66E+04	3.16E+01	5.14E+03
^{137}Cs	6.76E+04	2.68E+04	7.21E+03	5.20E+04
^{89}Sr	1.25E-01	3.47E-02	2.41E-35	1.01E-14
^{103}Ru	3.65E-03	1.13E-03	0.00E+00	5.47E-20
^{135}Cs	2.79E-01	1.11E-01	4.54E-02	2.16E-01
Fines				
^{225}Ac*	3.05E-08	2.14E-08	9.69E-09	9.89E-13
^{227}Ac*	2.36E-06	1.18E-06	1.45E-06	2.56E-08
^{110m}Ag	1.73E+02	6.58E+01	4.97E-06	2.04E-07
^{241}Am	4.76E+02	1.61E+02	2.52E+02	1.17E+00

Table 4.2.2 (continued) ISOTOPE INVENTORY Ci/Assembly				
	PWR MPCs-24 Ci/Assembly	MPC-68 Ci/Assembly	MPC-68F Ci/Assembly	Trojan MPCs Ci/Assembly
^{242M}Am*	5.60E+00	1.94E+00	9.35E-01	5.06E-03
^{243}Am*	2.23E+01	9.42E+00	3.30E+00	2.53E-02
^{137m}Ba	6.39E+04	2.53E+04	6.81E+03	0.00E+00
^{210M}Bi*	0.00E+00	0.00E+00	0.00E+00	1.38E-10
^{247}Bk*	2.82E-08	1.32E-08	5.94E-08	7.06E-24
^{144}Ce	4.77E+04	1.45E+04	7.33E-03	2.62E-04
^{248}Cf*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{249}Cf*	8.01E-05	4.47E-05	3.62E-06	7.20E-08
^{250}Cf*	2.92E-04	1.86E-04	6.69E-06	7.73E-08
^{251}Cf*	3.40E-06	2.06E-06	1.36E-07	2.84E-09
^{252}Cf*	4.11E-04	3.14E-04	3.64E-07	1.52E-08
^{254}Cf*	1.19E-13	1.05E-13	0.00E+00	5.32E-28
^{240}Cm*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{242}Cm*	3.21E+02	1.26E+02	7.71E-01	8.42E-05
^{243}Cm*	1.61E+01	6.51E+00	1.54E+00	9.51E-03
^{244}Cm	3.26E+03	1.43E+03	2.17E+02	1.42E+00
^{245}Cm*	3.25E-01	1.23E-01	2.48E-02	3.21E-04
^{246}Cm*	1.06E-01	5.40E-02	1.01E-02	1.14E-04
^{247}Cm*	7.07E-07	3.72E-07	5.26E-08	7.01E-10
^{248}Cm*	4.20E-06	2.43E-06	2.53E-07	1.56E-08

Table 4.2.2 (continued) ISOTOPE INVENTORY Ci/Assembly				
	PWR MPC-~~24~~*s* Ci/Assembly	MPC-68 Ci/Assembly	MPC-68F Ci/Assembly	Trojan MPCs Ci/Assembly
^{253}Es*	6.35E-20	4.62E-20	0.00E+00	0.00E+00
^{254}Es*	1.93E-08	1.96E-08	8.05E-16	5.24E-15
^{154}Eu	4.03E+03	1.47E+03	1.44E+02	1.01E-03
^{155}Eu	1.34E+03	5.46E+02	2.23E+01	6.06E-05
^{55}Fe	6.98E+01	3.23E+01	2.94E-01	1.11E-07
^{257}Fm*	4.26E-07	1.69E-07	0.00E+00	2.35E-26
^{148}Gd*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{182}Hf*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{236}Np*	9.77E-06	3.29E-06	7.30E-07	1.78E-09
^{237}Np*	2.33E-01	8.07E-02	2.55E-02	2.33E-04
^{239}Np	2.23E+01	9.42E+00	3.30E+00	1.01E-05
^{231}Pa*	1.82E-05	8.17E-06	3.16E-06	3.26E-08
^{210}Pb*	4.30E-09	2.17E-09	1.17E-08	3.77E-13
^{147}Pm	4.28E+04	1.52E+04	1.18E+02	2.17E-03
^{208}Po*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{209}Po*	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{210}Po*	3.92E-09	1.98E-09	1.08E-08	1.49E-13
^{144}Pr	4.77E+04	1.45E+04	7.33E-03	0.00E+00
^{144m}Pr	6.68E+02	2.04E+02	1.03E-04	0.00E+00
^{236}Pu*	2.04E-01	6.32E-02	3.66E-04	1.26E-05

Table 4.2.2 (continued) ISOTOPE INVENTORY Ci/Assembly				
	PWR MPCs-~~24~~ Ci/Assembly	MPC-68 Ci/Assembly	MPC-68F Ci/Assembly	Trojan MPCs Ci/Assembly
^{238}Pu	2.56E+03	9.55E+02	2.50E+02	2.37E+00
^{239}Pu	1.91E+02	6.24E+01	2.95E+01	2.00E-01
^{240}Pu	3.27E+02	1.34E+02	6.81E+01	3.70E-01
^{241}Pu	7.55E+04	2.47E+04	5.16E+03	1.21E+00
^{242}Pu*	1.65E+00	7.05E-01	3.06E-01	1.97E-03
^{244}Pu*	1.11E-13	6.58E-14	3.73E-14	2.87E-16
^{223}Ra*	2.37E-06	1.18E-06	1.45E-06	1.70E-11
^{225}Ra*	3.05E-08	2.14E-08	9.69E-09	4.94E-13
^{226}Ra*	2.82E-08	1.32E-08	5.94E-08	1.38E-12
^{106}Rh	4.97E+04	1.74E+04	2.30E-01	0.00E+00
^{222}Rn*	2.82E-08	1.32E-08	5.94E-08	6.89E-12
^{125}Sb	2.87E+03	1.15E+03	8.02E+00	1.59E-04
^{151}Sm	2.60E+02	7.92E+01	2.53E+01	1.24E-05
^{119m}Sn	5.46E+02	3.08E+02	1.07E-06	4.23E-05
^{125m}Te	6.99E+02	2.82E+02	1.96E+00	1.89E-03
^{227}Th*	2.33E-06	1.16E-06	1.43E-06	5.05E-11
^{228}Th*	8.56E-03	3.40E-03	1.71E-03	8.06E-06
^{229}Th*	3.05E-08	2.14E-08	9.69E-09	3.29E-10
^{230}Th*	2.16E-05	8.26E-06	1.29E-05	5.40E-08
^{230}U*	1.33E-23	4.74E-24	0.00E+00	0.00E+00

Table 4.2.2 (continued) ISOTOPE INVENTORY Ci/Assembly				
	PWR MPCs~~-24~~ Ci/Assembly	MPC-68 Ci/Assembly	MPC-68F Ci/Assembly	Trojan MPCs Ci/Assembly
^{232}U*	1.51E-02	5.58E-03	1.69E-03	1.21E-05
^{233}U*	1.41E-05	4.20E-06	3.03E-06	3.94E-09
^{234}U*	4.97E-01	1.70E-01	7.26E-02	1.08E-04
^{236}U*	1.60E-01	5.85E-02	1.84E-02	3.18E-05
^{90}Y	4.53E+04	1.76E+04	4.29E+03	4.13E-02

Note: The isotopes which contribute greater than 0.01% to the total curie inventory for the fuel assembly are considered in the evaluation as fines. Additionally, isotopes with A_2 values less than 1.0 in Table A-1, Appendix A, 10CFR71 are included as fines and are designated in the table by an "*".

Table 4.2.3

TOTAL SOURCE TERM FOR THE HI-STAR 100 SYSTEM (Ci/cm^3)

	C_{crud} (Ci/cm^3)	C_{fines} (Ci/cm^3)	C_{vol} (Ci/cm^3)	C_{gas} (Ci/cm^3)	Total (Ci/cm^3)
Normal Transport Conditions					
MPC-24	1.17E-05	1.26E-07	4.45E-06	1.60E-04	1.77E-04
MPC-24E, MPC-24EF	1.20E-05	1.29E-07	4.55E-06	1.64E-04	1.82E-04
Trojan MPC-24E	7.56E-06	5.31E-07	2.15E-06	1.04E-04	1.14E-04
Trojan MPC-24EF Secondary	7.77E-06	3.49E-06	1.42E-05	6.81E-04	7.06E-04
Trojan MPC-24EF Primary	7.56E-06	3.40E-06	1.38E-05	6.63E-04	6.88E-04
MPC-32	1.64E-05	1.77E-07	6.26E-06	2.25E-04	2.50E-04
MPC-68	1.08E-04	1.36E-07	5.20E-06	1.89E-04	3.03E-04
MPC-68F Secondary	2.00E-05	5.16E-07	2.28E-06	7.55E-05	9.83E-05
MPC-68F Primary	1.95E-05	5.02E-08	2.22E-06	7.35E-05	9.58E-05
Accident Conditions					
MPC-24	7.79E-05	4.20E-05	1.48E-04	5.34E-03	5.60E-03
MPC-24E, MPC-24EF	7.97E-05	4.29E-05	1.52E-04	5.46E-03	5.73E-03
Trojan MPC-24E	5.04E-05	1.77E-05	7.18E-05	3.45E-03	3.59E-03
Trojan MPC-24EF Secondary	5.18E-05	1.82E-05	7.37E-05	3.55E-03	3.69E-03
Trojan MPC-24EF Primary	5.04E-05	1.77E-05	7.18E-05	3.45E-03	3.59E-03
MPC-32	1.10E-04	5.90E-05	2.09E-04	7.51E-03	7.88E-03
MPC-68	7.18E-04	4.52E-05	1.73E-04	6.30E-03	7.23E-03
MPC-68F Secondary	1.34E-04	5.93E-06	2.62E-05	8.68E-04	1.03E-03
MPC-68F Primary	1.30E-04	5.77E-06	2.55E-05	8.45E-04	1.01E-03

Table 4.2.4

VARIABLES FOUND IN NUREG/CR-6487 USED IN THE LEAKAGE RATE ANALYSIS

Variable	PWR		BWR	
	Normal	Accident	Normal	Accident
Fraction of crud that spalls, f_C	0.15	1.0	0.15	1.0
Crud surface activity (Ci/cm^2)	140×10^{-06}	140×10^{-06}	1254×10^{-06}	1254×10^{-06}
Surface area per assembly, cm^2	3×10^5	3×10^5	1×10^5	1×10^5
Fraction of rods that develop cladding breach, f_B†	0.03	1.0	0.03	1.0
Fraction of fines that are released, f_f	3×10^{-5}	3×10^{-5}	3×10^{-5}	3×10^{-5}
Fraction of gases that are released, f_G	0.3	0.3	0.3	0.3
Fraction of volatiles that are released, f_V	2×10^{-04}	2×10^{-04}	2×10^{-04}	2×10^{-04}

† The calculation for normal transport conditions of the Trojan MPC-24EF and MPC-68F each containing four (4) DFCs with fuel debris assumes that for the four DFCs, 100% of the rods of the fuel debris are breached. The remaining 20 or 64 assemblies in the Trojan MPC-24EF and MPC-68F, respectively, were assumed to have a 3% cladding rupture. Therefore, f_B for the Trojan MPC-24EF and the MPC-68F containing fuel debris is 0.192 and 0.087, respectively.

Table 4.2.5

INDIVIDUAL CONTRIBUTOR EFFECTIVE A_2
FOR GASES, CRUD, FINES, AND VOLATILES

MPC Type	A_2 (Ci)
Gases	
PWR MPCs	282
MPC-68	282
MPC-68F	285
Trojan MPCs	478
Crud	
All MPCs	10.8
Fines	
PWR MPCs	0.308
MPC-68	0.284
MPC-68F	0.115
Trojan MPCs	0.147
Volatiles	
PWR MPCs	6.04
MPC-68	6.05
MPC-68F	5.43
Trojan MPCs	5.44

Table 4.2.6

TOTAL SOURCE TERM EFFECTIVE A_2 FOR NORMAL AND HYPOTHETICAL ACCIDENT CONDITIONS

Normal Transport Conditions	
	Effective A_2 (Ci)
MPC-24	27.4
MPC-24E MPC-24EF	27.4
Trojan MPC-24E	23.1
Trojan MPC-24EF	24.7
MPC-32	27.4
MPC-68	18.6
MPC-68F	14.0
Accident Conditions	
MPC-24	30.0
MPC-24E MPC-24EF	30.0
Trojan MPC-24E	24.6
Trojan MPC-24EF	24.6
MPC-32	30.0
MPC-68	26.2
MPC-68F	14.4

Table 4.2.7

RADIONUCLIDE RELEASE RATES

	Allowable Release Rate (R_N or R_A) (Ci/s)
Normal Conditions	
MPC-24	7.62E-09
MPC-24E, MPC-24EF	7.62E-09
Trojan MPC-24E	6.41E-09
Trojan MPC-24EF	6.87E-09
MPC-32	7.62E-09
MPC-68	5.18E-09
MPC-68F	3.88E-09
Accident Conditions	
MPC-24	4.94E-05
MPC-24E, MPC-24EF	4.94E-05
Trojan MPC-24E	4.06E-05
Trojan MPC-24EF	4.06E-05
MPC-32	4.94E-05
MPC-68	4.32E-05
MPC-68F	2.37E-05

Table 4.2.8

Table Deleted

Table 4.2.9

ALLOWABLE LEAKAGE RATES AT UPSTREAM PRESSURE

	C_{total} (Ci/cm^3)	Allowable Leakage Rate at P_u L_N or L_A (cm^3/s)
Normal Transport Conditions		
MPC-24	1.77E-04	4.29E-05
MPC-24E, MPC-24EF	1.82E-04	4.20E-05
Trojan MPC-24E	1.14E-04	5.63E-05
Trojan MPC-24EF Secondary	7.06E-04	9.73E-06
Trojan MPC-24EF Primary	6.88E-04	1.00E-05
MPC-32	2.50E-04	3.05E-05
MPC-68	3.03E-04	1.71E-05
MPC-68F Secondary	9.83E-05	3.95E-05
MPC-68F Primary	9.58E-05	4.05E-05
Accident Conditions		
MPC-24	5.60E-03	8.82E-03
MPC-24E, MPC-24EF	5.73E-03	8.62E-03
Trojan MPC-24E	3.59E-03	1.13E-02
Trojan MPC-24EF Secondary	3.69E-03	1.10E-02
Trojan MPC-24EF Primary	3.59E-03	1.13E-02
MPC-32	7.88E-03	6.27E-03
MPC-68	7.23E-03	5.96E-03
MPC-68F Secondary	1.03E-03	2.29E-02
MPC-68F Primary	1.01E-03	2.35E-02

Table 4.2.10

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Table 4.2.11

PLUTONIUM INVENTORY
(Ci/assembly)

Nuclide	MPC-68F MOX fuel Ci/Assy	MPC-68F UO_2 fuel Ci/Assy	Trojan MPC-24EF UO_2 fuel Ci/Assy
Pu-236	4.92E-04	3.66E-04	2.04E-01
Pu-237	0.00E+00	0.00E+00	3.04E-07
Pu-238	1.11E+03	2.50E+02	2.56E+03
Pu-239	3.29E+01	2.95E+01	1.91E+02
Pu-240	7.83E+01	6.81E+01	3.27E+02
Pu-241	6.15E+03	5.16E+03	7.55E+04
Pu-242	3.44E-01	3.06E-01	1.65E+00
Pu-244	0.0	3.73E-14	1.11E-13
Total	**7.37E+03**	**5.51E+03**	**7.86E+04**

Table 4.2.12

PARAMETERS FOR NORMAL, HYPOTHETICAL ACCIDENT AND TEST CONDITIONS

Parameter	Normal Conditions	Hypothetical Accident Conditions	Reference Test Conditions	Actual Test Conditions
P_u	104 psia[1] (7.07 ATM)	214.7 psia (14.61 ATM)	Primary: 1.68 ATM	Primary: 1.68 ATM (min)
			Secondary: 2.0 ATM	Secondary: 6.78 ATM (min)
P_d	14.7 psia (1 ATM)	14.7 psia (1 ATM)	14.7 psia (1 ATM)	14.7 psia (1 ATM)
T	495°F (530 K)	1058°F (843 K)	373 K	373 K (max)
M	4 g/mol	4 g/mol	4 g/mol	4 g/mol
u	0.0293 cP	0.0397 cP	0.0231 cP	0.0231 cP
a	Primary: 0.25 cm	Primary: 0.25 cm	Primary: 0.25 cm	Primary: 0.25 cm
	Secondary: 3.175 cm	Secondary: 3.175 cm	Secondary: 3.175 cm	Secondary: 3.175 cm

[1] The maximum upstream pressure for normal operating conditions in the Trojan MPCs is 83.2 psia (5.66 ATM). This value has been used to determine the maximum allowable leakage rate from the Trojan MPCs.

Table 4.2.13

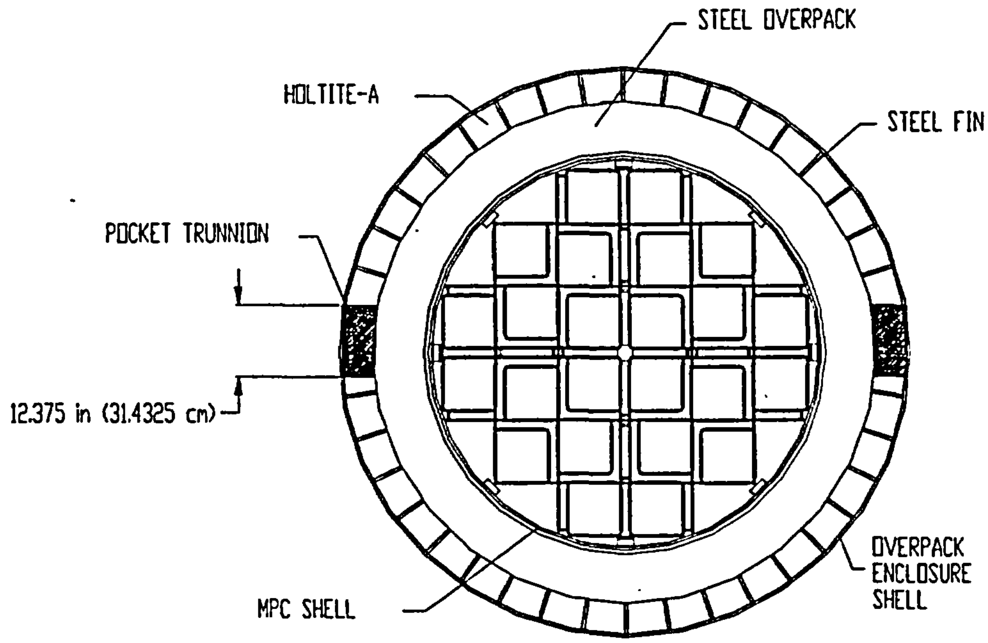
RADIONUCLIDE RELEASE RATES
FOR PLUTONIUM (SECONDARY CONTAINMENT)

	Effective A_2 (Ci)	Allowable Release Rate (R_N or R_A) (Ci/s)
Normal Transport Conditions		
MPC-68F MOX Fuel	0.0297	8.24E-12
MPC-68F UO_2 Fuel	0.0660	1.84E-11
Trojan MPC-24EF UO_2 Fuel	0.0926	2.57E-11
Accident Conditions		
MPC-68F	0.0297	4.89E-08
MPC-68F UO_2 Fuel	0.0660	1.09E-07
Trojan MPC-24EF UO_2 Fuel	0.0926	1.53E-07

Table 4.2.14

ALLOWABLE LEAKAGE RATES AT UPSTREAM PRESSURE FOR PLUTONIUM (SECONDARY CONTAINMENT)

	C_{Pu} (Ci/cm^3)	Allowable Leakage Rate at P_u L_N or L_A (cm^3/s)
Normal Transport Conditions		
MPC-68F MOX Fuel	2.18E-07	3.77E-05
MPC-68F UO_2 Fuel	1.63E-07	1.12E-04
Trojan MPC-24EF UO_2 Fuel	1.82E-06	1.41E-05
Accident Conditions		
MPC-68F	2.51E-06	1.95E-02
MPC-68F UO_2 Fuel	1.88E-06	5.81E-02
Trojan MPC-24EF UO_2 Fuel	9.49E-06	1.61E-02



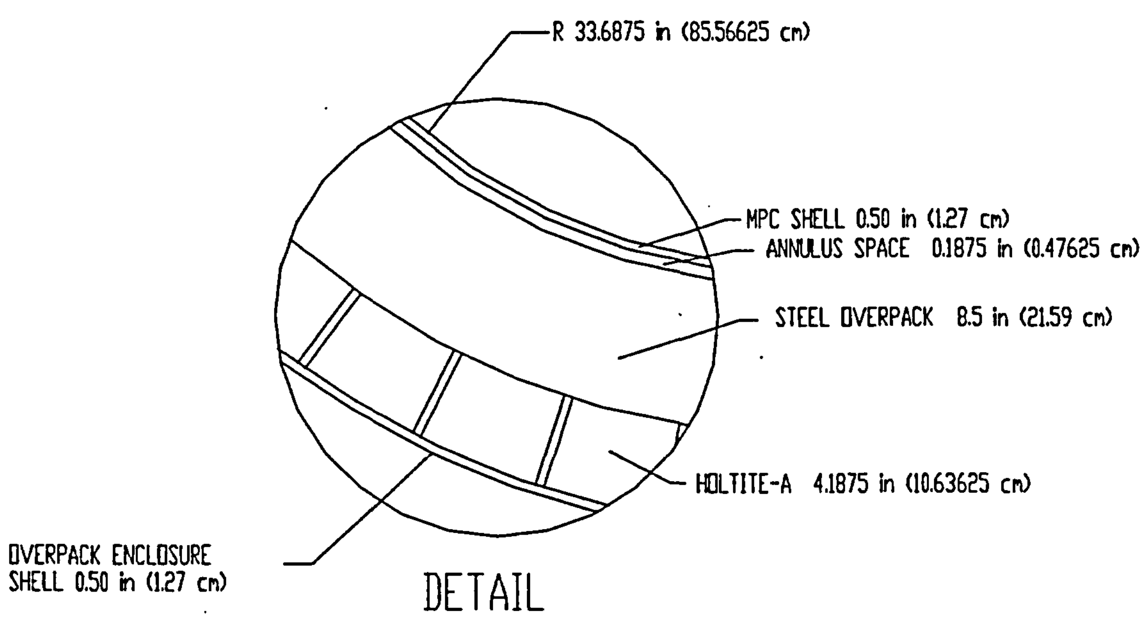


FIGURE 5.3.9; HI-STAR 100 OVERPACK CROSS SECTIONAL VIEW SHOWING THE THICKNESS OF THE MPC SHELL AND OVERPACK AS MODELED IN MCNP. THE MPC-24 IS SHOWN FOR ILLUSTRATIVE PURPOSES ONLY.

5.4 SHIELDING EVALUATION

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

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 1.2.15 and Figures 1.2.13 and 1.2.14. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6] respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions. The axial distribution used for the Trojan Plant fuel was similar but not identical to the generic PWR distribution. Table 1.2.15 and Figure 1.2.13a present the axial burnup distribution used for the Trojan Plant fuel taken from the Trojan FSAR [5.1.6].

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 1.2.15 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 1.2.15 for the generic 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 dose 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. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

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

Figures 5.1.1 and 5.1.2 depict the dose point locations during normal and hypothetical accident conditions of transport. Dose point location 3a in Figure 5.1.1 covers two regions of different radii. The outermost region is 5.75 inches in height and the innermost region is 6.875 inches in height. The dose rate was calculated over both segments and the highest value was reported for dose location 3a. Dose point locations 1 through 4 in Figure 5.1.2 are conservatively located at a radial position that is approximately 1 meter from the outer radial surface of the bottom plate.

Tables 5.4.8, 5.4.9, 5.4.19, 5.4.29, and 5.4.32 provide the total dose rate on the surface of the HI-STAR 100 System for each burnup level and cooling time. Tables 5.4.10 through 5.4.13, 5.4.20, 5.4.21, 5.4.30, 5.4.31, 5.4.33, and 5.4.34 provide the total dose rate at 2 meters for normal conditions and at 1 meter for accident conditions for each burnup level and cooling time for the MPC-24, MPC-68 and the MPC-32. This information was used to determine the worst case burnup level and cooling time and corresponding maximum dose rates reported in Section 5.1.

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 2% and the relative error for the individual dose components was typically less than 5%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions

The HI-STAR 100 overpack utilizes 0.5 inch thick radial channels for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through the neutron shield. Therefore, it is possible to have neutron streaming through the channels which could result in a localized dose peak. The reverse is true for photons which would result in a localized reduction in the photon dose. Analyses were performed to determine the magnitude of the dose peaks and depressions and the impact on localized dose as compared to average total dose. This effect was evaluated at the radial surface of the HI-STAR 100 System and a distance of two meters.

In addition to the radial channels, the pocket trunnions are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin. Therefore, analyses were performed to quantity this effect. Figures 5.3.7 and 5.3.8 illustrate the location of the pocket trunnion and its axial position relative to the active fuel.

The fuel loading pattern in the MPC-32, MPC-24 and the MPC-68, as depicted in Figures 5.3.1 through 5.3.3, is not cylindrical. Therefore, there is a potential to experience peaking as a result of azimuthal variations in the fuel loading. Since the MCNP models represent the fuel in the correct positions (i.e., cylindrical homogenization is not performed), the effect of azimuthal variations in the loading pattern is automatically accounted for in the calculations that are discussed below.

The effect of streaming through the pocket trunnion and the radial channels was analyzed using the full three-dimensional MCNP models of the MPC-24 and the MPC-68. The effect of peaking was calculated on the surface of the overpack adjacent to the pocket trunnion and dose locations 2a and 3a in Figures 5.1.1. The effect of peaking was also analyzed at 2 meters from the overpack at dose location 2 and at the axial height of the impact limiter. Dose location 3 was not analyzed at two meters because the dose at that point is less than the dose at location 2 as demonstrated in the tables at the end of this section. Figure 5.4.1 shows a quarter of the HI-STAR 100 overpack with 41 azimuthal bins drawn. There is one bin per steel fin and 3 bins in each neutron shield region. This azimuthal binning structure was used over the axial height of the overpack. The dose was calculated in each of these bins and then compared to the average dose calculated over the surface to determine a peak-to-average ratio for the dose in that bin. The azimuthal location of the pocket trunnion is shown in Figure 5.4.1. The pocket trunnion was modeled as solid steel. During shipping, a steel rotation trunnion or plug shall be placed in the pocket trunnion recess. To conservatively evaluate the peak to average ratio, the pocket trunnion is assumed to be solid steel.

Table 5.4.14 provides representative peak-to-average ratios that were calculated for the various dose components and locations. Table 5.4.15 presents the dose rates at the dose locations analyzed including the effect of peaking. These results can be compared with the surface average results in Tables 5.1.1, 5.1.3, 5.1.4, and 5.1.6. The peak dose on the surface of the overpack at dose location 2a occurs at a steel channel (fin). This is evident by the high neutron peaking at dose location 2a on the surface of the overpack. The dose rate at the pocket trunnion, in those overpacks containing pocket trunnions, is higher than the dose rate at dose location 2 on the surface of the overpack. However, these results clearly indicate that, at two meters, the peaking associated with the pocket trunnion is not present and that the peak dose location is #2.

The MPC-32 was not explicitly analyzed ~~for azimuthal peaking~~*to determine peak-to-average ratios. This is acceptable because the peaking outside the HI-STAR for* ~~It is expected that the peaking in~~ the MPC-32 will be similar if not smaller than in the MPC-24 due to the fact that the fuel assemblies in the MPC-24 are not as closely positioned to each other as in the MPC-32. *Section 5.5, Regulatory Compliance, presents results which take into account peaking due to* *radiation streaming or azimuthal variation. For the MPC-32, the peak-to-average values calculated for the MPC-24 were used.*

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

As discussed in Subsection 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of transporting intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies. As discussed in Subsection 5.2.8, the Trojan damaged fuel and fuel debris were not explicitly analyzed because they are bounded by the intact fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Section 1.4, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Some of the 6x6 assemblies described in Table 5.2.2 were manufactured with Inconel grid spacers (the mass of inconel is listed in Table 5.2.2). The calculated ^{60}Co activity from these spacers was 66.7 curies for a burnup of 30,000 MWD/MTU and a cooling time of 18 years. Including this source with the total fuel gamma source for damaged fuel in Table 5.2.6 and dividing by the 80 inch rubble height provides a gamma source per inch of 3.47E+12 photon/s. Dividing the total neutron source for damaged fuel in Table 5.2.14 by 80 inches provides a neutron source per inch of 3.93E+5 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 5.03E+12 photon/s and 6.63E+5 neutron/s. These BWR design basis values were calculated by dividing the total source strengths as calculated from Tables 5.2.5 and 5.2.13 (39,500 MWD/MTU and 14 year cooling values) by the active fuel length of 144 inches. Therefore, the design basis damaged fuel assembly is bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates are provided as they are bounded by the intact fuel analysis.

5.4.3 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden Unit 1 GE 6x6 MOX fuel assemblies can be compared to the design basis source terms for the BWR assemblies which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Including the ^{60}Co source from grid spacers as calculated in the previous subsection (66.7 curies) with the total fuel gamma source for the MOX fuel in Table 5.2.16 and dividing by the 110 inch active fuel height provides a gamma source per inch of 2.41E+12 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.17 by 110 inches provides a neutron source strength per inch of 3.67E+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 5.03E+12 photons/s and 6.63E+5 neutrons/s. These BWR design basis values were calculated by dividing the total source strengths as calculated from Tables 5.2.5 and 5.2.13 (39,500 MWD/MTU and 14 year cooling values) by the active fuel length of 144 inches. This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Subsection 5.4.2, the source term for the MOX fuel is calculated on a per inch basis assuming a post-accident rubble height of 80 inches. The resulting gamma and neutron source strengths are 3.31E+12 photons/s and 5.05E+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post-accident configuration.

5.4.4 Stainless Steel Clad Fuel Evaluation

Tables 5.4.22 through 5.4.24 present the dose rates from the stainless steel clad fuel at various dose locations around the HI-STAR 100 overpack for the MPC-24 and the MPC-68 for normal and hypothetical accident conditions. These dose rates are below the regulatory limits indicating that these fuel assemblies are acceptable for transport.

As described in Subsection 5.2.3, the source term for the stainless steel fuel was calculated conservatively with an artificial active fuel length of 144 inches. The end fitting masses of the stainless steel clad fuel are also assumed to be identical to the end fitting masses of the zircaloy clad fuel. In addition, the fuel assembly configuration used in the MCNP calculations was identical to the configuration used for the design basis fuel assemblies as described in Table 5.3.1.

5.4.5 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately 1E+8 neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as 5.8E-6 neutrons per gamma (1E+8/865/3.7E+10/0.54) with energy greater than 1.666 MeV or 1.16E+5 neutrons/curie (1E+8/865) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be 1.04E+8 gammas/sec which would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies. This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch (4.63E+6/77.25). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.25, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, transport of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

5.4.6 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.30 and 5.2.6 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.14, bounds the thoria rod neutron spectra, Table 5.2.31, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for transport in the MPC-68 or the MPC-68F.

5.4.7 Trojan Fuel Contents

Tables 5.4.26 through 5.4.28 present the results for the Trojan MPC-24E for normal surface and 2 meter as well as accident results. These results are presented for a single burnup and cooling time of 42,000 MWD/MTU and 16 year cooling. This burnup and cooling time combination is shown in Tables 5.2.33 through 5.2.35 to bound the other allowable burnup and cooling time combinations for Trojan fuel. Since the Trojan MPCs will contain BPRAs, RCCAs, and TPDs, the source from these devices was considered in the analysis. The source from BPRAs and TPDs were added to the fuel source in the appropriate location. The mass from these devices was conservatively neglected. Separate calculations were performed for the BPRAs and the TPDs since both devices can not be present in the same fuel assembly. The results presented in Tables 5.4.26 through 5.4.28 represent the configuration (fuel plus non-fuel hardware: BPRA or TPD) that produces the highest dose rate at that location. Separate results for the different non-fuel hardware are not provided. Separate MCNP calculations were performed for the consideration of the RCCAs since this source is localized at the bottom of the MPC. The results for the RCCAs indicate that the presence of RCCAs will increase the dose rate on the surface of the overpack by a maximum of 1.3 mrem/hr and the dose rate at 2 meters will increase by a maximum of 0.08 mrem/hr for normal conditions. During accident conditions the dose rate will increase by a maximum of 6 mrem/hr with the presence of RCCAs.

These dose rates are less than the regulatory limits and therefore the Trojan contents are approved for transportation.

5.4.8 Trojan Antimony-Beryllium Neutron Sources

The analysis of the Trojan secondary antimony-beryllium neutron sources was performed in a manner very similar to that described above in Subsection 5.4.5. The secondary sources are basically BPRAs with four rods containing the antimony-beryllium with a length of 88 inches in each rod. As mentioned in Subsection 5.4.5, the antimony-beryllium source is a regenerative source in which the antimony is activated and the gammas released from the antimony induce a gamma,n reaction in the beryllium.

The steady state production of neutrons from this antimony-beryllium source was conservatively calculated in the MPC using an approach very similar to that described in Subsection 5.4.5. The depletion of antimony from the operation in the reactor core was conservatively neglected in the analysis. MCNP calculations were performed with explicitly modeled fuel assemblies in a Trojan MPC model to calculate the steady state activity of Sb-124 in the antimony-beryllium source due to the neutrons from the spent fuel. This activity level was used in a subsequent MCNP calculation to determine the gamma,n reaction rate in the beryllium. The gamma,n cross section for beryllium, which exhibits peaks at 1.5E-3 with lows at approximately 0.3E-3 barns, was used in MCNP as a reaction rate multiplier for the flux tallies. Additionally, the gamma,n reaction rate due to gammas from the spent fuel was determined. In the latter case, gammas from the spent fuel with energies up to 11 MeV were considered in the analysis compared to an upper limit of 3 MeV for the cask dose rate analysis. Finally, the gamma,n reaction rate was converted to neutrons/sec to yield the neutron source per secondary source assembly. In this conversion process the spectrum of neutrons emitted from the Sb-Be source was determined based on the energy spectrum of the gammas reacting in the beryllium [5.4.7]. The neutron source strength per secondary source assembly was calculated to be 9.9E+5 neutrons/sec with more than 99% of these having an upper energy of 0.03 MeV. The remaining 1% of the secondary source neutrons had energies up to 0.74 MeV. This is a conservative estimate of the neutrons/sec from the secondary source because it neglects depletion of the antimony that has occurred during core operation and it assumes that all assemblies in the MPC are design basis Trojan fuel assemblies.

In order to determine the impact of the secondary neutron sources on the dose rates, MCNP calculations were performed. Since the dose rate that is closest to the regulatory limit is at 2 meters from the overpack, this was the only location considered in the analysis. Rather than calculate the average dose rate around the overpack at the 2 meter location, the dose rate was calculated for a specific location. Figure 5.4.2 shows the location where the dose rate was calculated. This location (an 8.2 inch diameter cylinder) is at 2 meters from the transport vehicle on a line drawn from the center of the MPC through the center of a corner assembly. The dose rate in this cylinder was calculated using the same axial segmentation as in the design basis calculations. In this analysis, the corner assembly was the only assembly considered to have the secondary source assembly. This choice of assembly position and dose location bounds all other possible locations for the single Trojan secondary source assembly permitted in any MPC.

The dose rates were calculated for the following combinations of fuel assemblies and non-fuel hardware inserts. In all dose rate calculations, both the neutron and gamma source from the secondary sources was considered.

1. One fuel assembly with secondary source assembly from cycles 1-4 and the remaining 23 fuel assemblies with BPRAs.
2. One fuel assembly with secondary source assembly from cycles 1-4 and the remaining 23 fuel assemblies with TPDs.
3. One fuel assembly with secondary source assembly from cycles 4-14 and the remaining 23 fuel assemblies with BPRAs.
4. One fuel assembly with secondary source assembly from cycles 4-14 and the remaining 23 fuel assemblies with TPDs.

The worst case dose rate from the configurations listed above was less than 9.8 mrem/hr from configuration 4. This value was conservatively calculated assuming all fuel assemblies were identical design basis Trojan fuel assemblies with design basis Trojan non-fuel hardware. This dose rate is slightly higher than the design basis dose rates for the Trojan fuel. However, this value is still below the regulatory limit of 10.0 mrem/hr. Therefore, the insertion of a single secondary source assembly into a Trojan MPC is acceptable for transport.

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

DELETED

Table 5.4.3

DELETED

Table 5.4.4

DELETED

Table 5.4.5

DELETED

Table 5.4.6

DELETED

Table 5.4.7

DELETED

Table 5.4.8

TOTAL DOSE RATES
DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 6 Year Cooling (mrem/hr)	29,500 MWD/MTU 7 Year Cooling (mrem/hr)	34,500 MWD/MTU 9 Year Cooling (mrem/hr)	39,500 MWD/MTU 11 Year Cooling (mrem/hr)	44,500 MWD/MTU 14 Year Cooling (mrem/hr)
2a	49.81	50.88	46.38	43.02	46.19
3a	95.80	108.16	113.72	124.43	138.47
1	35.33	37.42	36.18	35.89	34.85
2	29.01	28.87	26.11	26.57	28.24
3	27.02	29.30	29.26	29.94	30.19
4	23.73	26.05	26.43	27.40	28.05
5	1.04	1.70	2.51	3.36	4.33
6	120.60	122.66	111.08	102.27	89.54
10CFR71.47 Limit	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)

† Refer to Figure 5.1.1.

Table 5.4.9

TOTAL DOSE RATES
DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 11 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
2a	44.53	44.60	50.12	52.65	55.25
3a	132.09	129.62	132.20	115.66	103.70
1	34.23	35.33	37.96	35.72	33.64
2	24.21	27.97	31.12	32.05	33.07
3	32.25	31.43	31.69	27.33	22.81
4	30.52	29.69	29.91	25.74	21.42
5	0.71	1.16	1.78	2.28	2.86
6	99.53	95.87	95.27	80.46	64.92
10CFR71.47 Limit	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)

† Refer to Figure 5.1.1.

Table 5.4.10

TOTAL DOSE RATES
DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 6 Year Cooling (mrem/hr)	29,500 MWD/MTU 7 Year Cooling (mrem/hr)	34,500 MWD/MTU 9 Year Cooling (mrem/hr)	39,500 MWD/MTU 11 Year Cooling (mrem/hr)	44,500 MWD/MTU 14 Year Cooling, (mrem/hr)
1	7.41	7.61	7.26	7.31	7.27
2	9.57	9.45	8.77	8.95	9.10
3	6.72	6.93	6.61	6.64	6.59
4	6.16	6.39	6.13	6.16	6.11
5	0.10	0.17	0.24	0.32	0.41
6	8.11	8.01	6.91	5.99	4.75
10CFR71.47 Limit	10.00	10.00	10.00	10.00	10.00

† Refer to Figure 5.1.1.

Table 5.4.11

TOTAL DOSE RATES
DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 11 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
1	6.58	7.05	7.59	7.34	7.07
2	8.03	8.94	9.62	9.55	9.39
3	6.31	6.44	6.62	6.02	5.36
4	6.09	6.15	6.32	5.70	5.03
5	0.08	0.13	0.20	0.26	0.32
6	5.67	5.25	4.94	3.84	2.65
10CFR71.47 Limit	10.00	10.00	10.00	10.00	10.00

† Refer to Figure 5.1.1.

Table 5.4.12

TOTAL DOSE RATES
DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	**24,500 MWD/MTU 6 Year Cooling (mrem/hr)**	**29,500 MWD/MTU 7 Year Cooling (mrem/hr)**	**34,500 MWD/MTU 9 Year Cooling (mrem/hr)**	**39,500 MWD/MTU 11 Year Cooling (mrem/hr)**	**44,500 MWD/MTU 14 Year Cooling (mrem/hr)**
1	76.55	97.55	117.95	141.21	166.93
2	153.26	224.23	307.31	399.33	504.35
3	49.37	64.41	79.67	96.85	116.00
4	35.97	47.12	58.44	71.10	85.20
5	4.06	6.59	9.62	12.83	16.51
6	685.36	687.08	605.96	539.45	447.78
10CFR71.51 Limit	1000.00	1000.00	1000.00	1000.00	1000.00

† Refer to Figure 5.1.2.

Table 5.4.13

TOTAL DOSE RATES
DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 11 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
1	82.43	108.56	145.23	169.25	198.52
2	179.75	275.87	403.87	504.00	622.86
3	46.51	59.30	77.64	88.84	102.65
4	36.57	45.32	58.13	65.34	74.36
5	2.84	4.55	6.92	8.80	11.04
6	564.75	530.59	509.65	408.69	300.71
10CFR71.51 Limit	1000.00	1000.00	1000.00	1000.00	1000.00

† Refer to Figure 5.1.2.

Table 5.4.14

PEAK-TO-AVERAGE RATIOS FOR THE DOSE COMPONENTS
AT VARIOUS LOCATIONS

Location	Fuel Gammas	Gammas from Neutrons	^{60}Co Gammas	Neutron
MPC-24				
Surface				
Pocket Trunnion	0.081	0.262	0.075	6.695
2a	0.713	0.955	0.407	2.362
3a	1.317	1.011	1.005	1.177
2 meter				
Pocket Trunnion	1.109	1.232	1.059	0.809
2	1.034	0.974	1.086	0.990
MPC-68				
Surface				
Pocket Trunnion	0.070	0.432	0.074	7.340
2a	0.737	0.977	1.123	2.284
3a	0.908	0.816	1.217	0.940
2 meter				
Pocket Trunnion	1.121	0.982	1.144	1.171
2	1.070	0.939	1.146	0.950

Table 5.4.15

DOSE RATES FOR NORMAL CONDITIONS SHOWING THE EFFECT OF PEAKING

Dose Point† Location	Fuel Gammas (mrem/hr)	Gammas from Neutrons (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Total (mrem/hr)
MPC-24					
Surface 44,500 MWD/MTU 14-Year Cooling					
Pocket Trunnion	0.15	0.37	1.98	97.92	100.42
2a	12.30	6.35	0.00	52.60	71.26
3a	0.40	0.67	28.67	128.27	158.01
2 meter 24,500 MWD/MTU 6-Year Cooling					
Pocket Trunnion	4.03	0.17	3.50	0.64	8.34
2	7.55	0.21	1.26	0.87	9.90
MPC-68					
Surface 34,500 MWD/MTU 11-Year Cooling					
Pocket Trunnion	0.25	0.45	1.97	77.42	80.09
2a	19.24	5.35	0.02	42.33	66.93
3a	0.33	0.12	115.34	34.69	150.49
2 meter 34,500 MWD/MTU 11-Year Cooling					
Pocket Trunnion	3.23	0.46	2.06	3.03	8.77
2	5.80	0.68	0.74	2.69	9.91

† Refer to Figure 5.1.1.

Table 5.4.16

DELETED

Table 5.4.17

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Table 5.4.18

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Table 5.4.19

TOTAL DOSE RATES
DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 9 Year Cooling (mrem/hr)	29,500 MWD/MTU 11 Year Cooling (mrem/hr)	34,500 MWD/MTU 13 Year Cooling (mrem/hr)	39,500 MWD/MTU 15 Year Cooling (mrem/hr)	44,500 MWD/MTU 18 Year Cooling (mrem/hr)
2a	42.91	42.11	43.11	45.08	46.24
3a	69.70	74.06	83.00	96.79	111.87
1	25.77	25.19	25.54	26.31	26.75
2	26.97	26.54	27.34	28.35	28.62
3	19.70	19.86	20.87	22.23	23.47
4	17.23	17.64	18.84	20.37	21.85
5	0.92	1.46	2.16	2.89	3.73
6	84.38	77.61	72.77	69.20	63.58
10CFR71.47 Limit	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)

† Refer to Figure 5.1.1.

Table 5.4.20

TOTAL DOSE RATES
DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	**24,500 MWD/MTU 9 Year Cooling (mrem/hr)**	**29,500 MWD/MTU 11 Year Cooling (mrem/hr)**	**34,500 MWD/MTU 13 Year Cooling (mrem/hr)**	**39,500 MWD/MTU 15 Year Cooling (mrem/hr)**	**44,500 MWD/MTU 18 Year Cooling (mrem/hr)**
1	6.42	6.24	6.29	6.42	6.40
2	9.51	9.16	9.18	9.27	9.09
3	5.68	5.52	5.58	5.70	5.71
4	5.08	4.96	5.03	5.16	5.19
5	0.09	0.14	0.21	0.28	0.35
6	5.67	5.00	4.40	3.88	3.17
10CFR71.47 Limit	10.00	10.00	10.00	10.00	10.00

† Refer to Figure 5.1.1.

Table 5.4.21

TOTAL DOSE RATES
DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 9 Year Cooling (mrem/hr)	29,500 MWD/MTU 11 Year Cooling (mrem/hr)	34,500 MWD/MTU 13 Year Cooling (mrem/hr)	39,500 MWD/MTU 15 Year Cooling (mrem/hr)	44,500 MWD/MTU 18 Year Cooling (mrem/hr)
1	62.33	76.36	95.99	117.23	140.85
2	145.00	201.06	275.44	354.15	442.83
3	40.70	51.15	65.54	81.04	98.36
4	29.39	37.13	47.76	59.20	72.01
5	3.59	5.63	8.26	11.03	14.21
6	478.28	429.22	388.56	354.69	306.90
10CFR71.51 Limit	1000.00	1000.00	1000.00	1000.00	1000.00

† Refer to Figure 5.1.2.

Table 5.4.22

DOSE RATES FOR
MPC-68 DESIGN BASIS STAINLESS STEEL CLAD FUEL
22,500 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Dose Location at Surface for Normal Condition				
1	2.91	9.19	1.00	13.09
2a	39.68	0.00	1.20	40.88
3a	0.62	40.84	2.60	44.07
4	0.45	9.49	0.53	10.47
5	0.01	0.01	0.11	0.14
6	2.35	31.19	1.40	34.93
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	3.45	1.00	0.17	4.63
2	7.71	0.27	0.19	8.18
3	2.26	1.35	0.12	3.73
4	1.67	1.43	0.11	3.21
5	0.00	0.00	0.01	0.02
6	0.20	1.82	0.03	2.05
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	9.43	10.90	7.95	28.29
2	46.22	0.23	25.97	72.42
3	3.58	7.41	4.06	15.05
4	2.00	6.60	2.91	11.51
5	0.01	0.07	0.48	0.57
6	11.14	183.23	5.34	199.71
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.23

DOSE RATES FOR
MPC-24 DESIGN BASIS STAINLESS STEEL CLAD FUEL
30,000 MWD/MTU AND 19-YEAR COOLING

Dose Point[†] Location	Fuel Gammas[††] (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Dose Location at Surface for Normal Condition				
1	2.40	5.54	4.27	12.22
2a	35.54	0.01	4.41	39.96
3a	0.66	11.31	24.60	36.57
4	0.73	3.65	4.11	8.49
5	0.11	0.01	0.86	0.98
6	4.19	21.13	7.14	32.46
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	3.05	0.69	0.76	4.50
2	7.23	0.23	0.83	8.29
3	2.47	0.66	0.72	3.85
4	1.95	0.67	0.69	3.30
5	0.01	0.00	0.08	0.09
6	0.36	1.48	0.18	2.02
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	7.57	6.83	32.78	47.18
2	39.78	0.24	108.52	148.54
3	4.96	3.96	23.27	32.19
4	2.85	3.00	17.13	22.99
5	0.02	0.05	3.69	3.76
6	22.73	123.24	28.03	174.00
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

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† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.24

DOSE RATES FOR
MPC-24 DESIGN BASIS STAINLESS STEEL CLAD FUEL
40,000 MWD/MTU AND 24-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Dose Location at Surface for Normal Condition				
1	2.12	5.80	11.10	19.02
2a	28.04	0.00	13.06	41.10
3a	0.78	11.82	63.88	76.48
4	0.66	3.82	10.68	15.16
5	0.29	0.01	2.24	2.53
6	4.28	22.10	18.54	44.92
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	2.55	0.72	1.98	5.26
2	5.82	0.24	2.23	8.29
3	2.06	0.69	1.86	4.62
4	1.64	0.70	1.78	4.11
5	0.02	0.00	0.22	0.24
6	0.29	1.55	0.47	2.31
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	5.88	7.14	85.12	98.14
2	30.69	0.25	281.83	312.76
3	3.85	4.14	60.42	68.41
4	2.24	3.14	44.48	49.86
5	0.04	0.05	9.57	9.66
6	17.44	128.89	72.74	219.07
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.25

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

Assembly	**Active fuel length (inch)**	**Neutrons per sec per inch**	**Neutrons per sec per inch with Sb-Be source**	**Reference for neutrons per sec per inch**
7x7 design basis	144	6.63E+5	N/A	Table 5.2.13 39.5 GWD/MTU and 14 year cooling
6x6 design basis	110	2.85E+5	3.45E+5	Table 5.2.14
6x6 design basis MOX	110	3.67E+5	4.27E+5	Table 5.2.17

Table 5.4.26

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
42,000 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	10 CFR 71.47 Limit
2a	3.72	2.48	38.39	2.25	46.84	1000
3a	0.39	0.07	14.34	49.81	64.61	1000
1	1.62	1.09	4.54	14.43	21.68	200
2	10.94	7.72	0.05	9.69	28.40	200
3	0.62	0.32	10.66	8.00	19.60	200
4	0.36	0.16	5.01	7.80	13.34	200
5	0.34†††	-	0.06	3.17	3.58	200
6	6.99†††	-	21.45	23.26	51.70	200

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.27

DOSE RATES AT TWO METERS FROM THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
42,000 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	1.52	1.11	0.55	2.52	5.69
2	3.41	2.53	0.64	2.67	9.24
3	1.20	0.82	2.31	1.71	6.05
4	0.97	0.62	2.13	1.50	5.21
5	0.02†††	-	0.05	0.27	0.34
6	0.56†††	-	2.05	0.87	3.49
10CFR71.47 Limit					10.00

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.28

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON- ZIRCALOY INCORE SPACERS
42,000 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	⁶⁰Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	7.01	5.69	106.18	118.88
2	31.31	0.30	356.39	387.99
3	3.27	15.94	69.43	88.64
4	1.86	8.65	49.17	59.68
5	0.11	0.25	12.42	12.78
6	34.74	128.20	82.48	245.42
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.29

TOTAL DOSE RATES
DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 12 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
2a	62.31	66.79	58.99	59.34	50.90
3a	162.02	192.08	205.87	245.88	263.95
1	44.46	48.87	45.01	46.94	42.51
2	34.63	38.55	36.17	38.33	35.48
3	40.30	46.47	46.08	50.81	49.68
4	37.52	43.47	43.40	48.09	47.31
5	2.40	3.97	5.66	7.60	9.11
6	144.35	150.66	126.87	122.14	97.35
10CFR71.47 Limit	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)

† Refer to Figure 5.1.1.

Table 5.4.30

TOTAL DOSE RATES
DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 12 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
1	7.85	8.67	8.01	8.43	7.69
2	8.22	9.28	8.86	9.65	9.19
3	7.83	8.83	8.44	9.10	8.59
4	7.50	8.48	8.14	8.79	8.34
5	0.22	0.37	0.52	0.70	0.83
6	7.98	8.02	6.27	5.60	3.82
10CFR71.47 Limit	10.00	10.00	10.00	10.00	10.00

† Refer to Figure 5.1.1.

Table 5.4.31

TOTAL DOSE RATES
DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 8 Year Cooling (mrem/hr)	29,500 MWD/MTU 9 Year Cooling (mrem/hr)	34,500 MWD/MTU 12 Year Cooling (mrem/hr)	39,500 MWD/MTU 14 Year Cooling (mrem/hr)	44,500 MWD/MTU 19 Year Cooling (mrem/hr)
1	91.38	117.99	134.85	162.88	176.78
2	150.55	230.95	310.10	406.55	477.18
3	66.27	89.91	108.35	134.66	150.49
4	49.53	66.55	79.40	98.17	109.14
5	9.17	15.09	21.41	28.69	34.38
6	802.16	817.16	656.05	601.72	437.06
10CFR71.51 Limit	1000.00	1000.00	1000.00	1000.00	1000.00

† Refer to Figure 5.1.2.

Table 5.4.32

TOTAL DOSE RATES
DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 12 Year Cooling (mrem/hr)	29,500 MWD/MTU 14 Year Cooling (mrem/hr)	34,500 MWD/MTU 16 Year Cooling (mrem/hr)	39,500 MWD/MTU 19 Year Cooling (mrem/hr)	42,500 MWD/MTU 20 Year Cooling (mrem/hr)
2a	40.66	40.41	42.78	42.27	44.54
3a	110.58	127.81	162.16	189.57	213.21
1	29.82	30.69	32.88	33.09	35.33
2	24.88	25.46	27.44	27.93	29.93
3	27.57	30.30	34.70	37.26	40.74
4	25.39	28.15	32.50	35.17	38.57
5	2.07	3.30	4.88	6.32	7.32
6	91.55	87.51	86.14	78.83	81.03
10CFR71.47 Limit	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)	1000.00 (2a,3a) 200.00 (1-6)

† Refer to Figure 5.1.1.

Table 5.4.33

TOTAL DOSE RATES
DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point† Location	24,500 MWD/MTU 12 Year Cooling (mrem/hr)	29,500 MWD/MTU 14 Year Cooling (mrem/hr)	34,500 MWD/MTU 16 Year Cooling (mrem/hr)	39,500 MWD/MTU 19 Year Cooling (mrem/hr)	42,500 MWD/MTU 20 Year Cooling (mrem/hr)
1	6.55	6.64	6.95	6.78	7.13
2	8.99	9.14	9.54	9.21	9.61
3	6.38	6.65	7.17	7.23	7.71
4	5.93	6.21	6.76	6.88	7.36
5	0.19	0.30	0.45	0.58	0.67
6	4.95	4.46	4.05	3.32	3.24
10CFR71.47 Limit	10.00	10.00	10.00	10.00	10.00

† Refer to Figure 5.1.1.

Table 5.4.34

TOTAL DOSE RATES
DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT VARYING BURNUPS AND COOLING TIMES

Dose Point[†] Location	**24,500 MWD/MTU 12 Year Cooling (mrem/hr)**	**29,500 MWD/MTU 14 Year Cooling (mrem/hr)**	**34,500 MWD/MTU 16 Year Cooling (mrem/hr)**	**39,500 MWD/MTU 19 Year Cooling (mrem/hr)**	**42,500 MWD/MTU 20 Year Cooling (mrem/hr)**
1	69.80	86.91	110.36	128.98	144.97
2	143.48	202.73	279.80	346.19	395.59
3	52.15	68.42	90.21	108.35	122.82
4	38.31	49.93	65.54	78.49	88.90
5	7.89	12.51	18.46	23.84	27.60
6	500.54	460.23	430.39	368.33	367.17
10CFR71.51 Limit	1000.00	1000.00	1000.00	1000.00	1000.00

† Refer to Figure 5.1.2.

CHAPTER 6: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STAR 100 System for the packaging and transportation of radioactive materials (spent nuclear fuel) in accordance with 10CFR71. The results of this evaluation demonstrate that, for the designated fuel assembly classes and basket configurations, an infinite number of HI-STAR 100 Systems with variations in internal and external moderation remain subcritical with a margin of subcriticality greater than 0.05Δk. This corresponds to a transport index of zero (0) and demonstrates compliance with 10CFR71 criticality requirements for normal and hypothetical accident conditions of transport.

The criticality design is based on favorable geometry, fixed neutron poisons (Boral), an administrative limit on the maximum allowable enrichment, and an administrative limit on the minimum average assembly burnup for the MPC-32. Criticality safety of the HI-STAR 100 System does *not* rely on credit for: (1) fuel burnup except for the MPC-32; (2) fuel-related burnable absorbers; or (3) more than 75% of the manufacturer's minimum B-10 content for the Boral neutron absorber.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STAR 100 System design structures and components important to criticality safety and limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package.

~~Note:~~
~~The MPC-32 requires burnup credit. Methodology and results for burnup credit are not yet presented in this chapter, and will be added as Appendix 6.E in a later revision of this SAR. However, general discussions regarding the MPC-32, tables for burnup credit results and references to Appendix 6.E have already been added throughout this chapter.~~

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in 10CFR71 [6.1.1], NUREG-1617 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STAR 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal and hypothetical accident conditions of transport. This criterion provides a large subcritical margin, sufficient to assure the criticality safety of the HI-STAR 100 System when fully loaded with fuel of the highest permissible reactivity. In addition, the results of this evaluation demonstrate that the HI-STAR 100 System is in full compliance with the requirements outlined in the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536.

Criticality safety of the HI-STAR 100 System depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24),

2. The incorporation of permanent fixed neutron-absorbing panels (Boral) in the fuel basket structure, and

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

4. An administrative limit on the minimum average assembly burnup for PWR fuel in the MPC-32.

The HI-STAR 100 System is designed such that the fixed neutron absorber (Boral) will remain effective for a period greater than 20 years, and there are no credible means to lose it. Therefore, 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-STAR 100 System does not rely on the use of any of the following credits:

- burnup of fuel, except for the MPC-32

- fuel-related burnable neutron absorbers

- more than 75 percent of the B-10 content for the fixed neutron absorber (Boral).

The following interchangeable basket designs are available for use in the HI-STAR 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment.

- a 24-cell basket (MPC-24E/EF), designed for intact and damaged PWR fuel assemblies, and fuel debris (MPC-24EF only). This is a variation of the MPC-24, with increased ^{10}B content in the Boral and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC). The MPC-24E and MPC-24EF is designed for fuel assemblies with a specified maximum enrichment. Although the MPC-24E/EF is designed and analyzed for damaged fuel and fuel debris, it is only certified for intact fuel assemblies.

- a 24-cell basket (MPC-24E/EF Trojan), design for intact and damaged PWR fuel assemblies, and fuel debris (MPC-24EF Trojan only) from the Trojan Nuclear Plant (TNP). This is a variation of the MPC-24E/EF, with a slightly reduced height, and increased cell sizes for the cells designated for damaged fuel and fuel debris. This increased cell size is required to accommodate the Trojan specific Failed Fuel Cans and DFCs.

- a 32-cell basket (MPC-32), designed for intact PWR fuel assemblies of a specified minimum burnup, and

- a 68-cell basket (MPC-68), designed for both intact and damaged BWR fuel assemblies with a specified maximum planar-average enrichment. Additionally, a variation in the MPC-68, designated MPC-68F, is designed for damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.

During the normal conditions of transport, the HI-STAR 100 System is dry (no moderator), and thus, the reactivity is very low ($k_{eff} < 0.50$). However, the HI-STAR 100 System for loading and unloading operations, as well as for the hypothetical accident conditions, is flooded, and thus, represents the limiting case in terms of reactivity. The calculational models for these conditions conservatively include: full flooding with ordinary water, corresponding to the highest reactivity, and the worst case (most conservative) combination of manufacturing and fabrication tolerances.

The MPC-24EF contains the same basket as the MPC-24E. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF. Therefore, all criticality results obtained for the MPC-24E are valid for the MPC-24EF and no separate analyses for the MPC-24EF are necessary.

Confirmation of the criticality safety of the HI-STAR 100 Systems under flooded conditions, when filled with fuel of the maximum permissible reactivity for which they are designed, was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

For the burnup credit calculations, CASMO-4, a two-dimensional transport theory code [6.1.10-6.1.12] for fuel assemblies, was used to calculate the isotopic composition of the spent fuel. The criticality evaluations for burnup credit were performed with MCNP4a [6.1.4].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO and MCNP4a [6.1.4] were used. The CASMO and MCNP4a calculations identify those tolerances that cause a positive reactivity effect, enabling the Monte Carlo code input to define the worst case (most conservative) conditions. CASMO was not used for quantitative criticality evaluations, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a, CASMO and KENO5a) with experimental data, using experiments selected to encompass, insofar as practical, the design parameters of the HI-STAR 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24) or cell spacing (MPC-32 and MPC-68), (3) the ^{10}B loading of the neutron absorber panels, and (4) the assembly burnup (MPC-32 only). Benchmark calculations are presented in Appendix 6.A and Appendix 6.E.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- U.S. Code of Federal Regulations, "Packaging and Transportation of Radioactive Materials," Title 10, Part 71.

- NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" USNRC, Washington D.C., March 2000.

- U.S. Code of Federal Regulations, "Prevention of Criticality in Fuel Storage and Handling," Title 10, Part 50, Appendix A, General Design Criterion 62.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

- USNRC Interim Staff Guidance 8 (ISG-8), Revision 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks".

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

- The MPCs are assumed to contain the most reactive 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 poisons, except for fuel in the MPC-32.

- The criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber.

- The fuel stack density is assumed to be 96% of theoretical (10.522 g/cm^3) for all criticality analyses. The fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels might be slightly greater than 96% of theoretical, the actual stack density will still be less.

- For fresh fuel, no credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be water at a temperature corresponding to the highest reactivity within the expected operating range (i.e., water density of 1.000 g/cc).

- Neutron absorption in minor structural members and optional heat conduction elements is neglected, i.e., spacer grids, basket supports, and optional aluminum heat conduction elements are replaced by water.

- 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.

- Planar-averaged enrichments are assumed for BWR fuel. (Analyses are presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)

- 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 reactivity bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000.

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

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

- For intact fuel assemblies, as defined in Chapter 1, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

- The burnup credit methodology for the MPC-32 contains significant additional conservative assumption specific to burnup credit, as discussed in Appendix 6.E.

The principal calculational results, which address the following conditions:

- A single package, under the conditions of 10 CFR 71.55(b), (d), and (e);
- An array of undamaged packages, under the conditions of 10 CFR 71.59(a)(1); and
- An array of damaged packages, under the conditions of 10 CFR 71.59(a)(2)

are summarized in Table 6.1.4 for all MPCs and for the most reactive configuration and fuel condition in each MPC. These results demonstrate that the HI-STAR 100 System is in full compliance with 10CFR71 (71.55(b), (d), and (e) and 71.59(a)(1) and (a)(2)). The calculations for package arrays are performed for infinite arrays of HI-STAR 100 Systems under flooded conditions. Therefore, the transportation index based on criticality control is zero (0). It is noted that the results for the internally flooded single package and package arrays are statistically equivalent for each basket. This shows that the physical separation between overpacks and the steel radiation shielding are each adequate to preclude any significant neutronic coupling between casks in an array configuration. In addition, the table shows the result for an unreflected, internally flooded cask for each MPC. This configuration is used in many calculations and studies throughout this chapter, and is shown to yield results that are statistically equivalent to the results for the corresponding reflected package. Further 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. These analyses also include cases with various internal and external moderator densities and various cask-to-cask spacings.

Additional results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) are listed in Tables 6.1.1 through 6.1.3 and 6.1.5 through 6.1.7, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs and fuel assembly classes†, Tables 6.1.1 through 6.1.3 and 6.1.5 through 6.1.7 list the bounding maximum k_{eff} value, the associated maximum allowable enrichment, and the minimum required assembly average burnup (if applicable), as required by 10CFR71.33(b)(2). The maximum enrichment and minimum burnup acceptance criteria are defined in Chapter 1. Additional results for each of the candidate fuel assemblies, that are bounded by those listed in Tables 6.1.1 through 6.1.3, are given in Section 6.2 for the MPC-24, MPC-68 and MPC-68F. The tables in Section 6.2 list the maximum k_{eff} (including bias, uncertainties, and calculational statistics), calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations analyzed. The capability of the MPC-68F to safely accommodate Dresden-1 and Humboldt Bay damaged fuel (fuel assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) is demonstrated in Subsection 6.4.4.

In Summary, these results confirm that the maximum k_{eff} values for the HI-STAR 100 System are below the limiting design criteria (k_{eff} < 0.95) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. The transportation index based on criticality control is zero (0).

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

Table 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}
14x14A	4.6	0.9296
14x14B	4.6	0.9228
14x14C	4.6	0.9307
14x14D	4.0	0.8507
14x14E	5.0	0.7627
15x15A	4.1	0.9227
15x15B	4.1	0.9388
15x15C	4.1	0.9361
15x15D	4.1	0.9367
15x15E	4.1	0.9392
15x15F	4.1	0.9410
15x15G	4.0	0.8907
15x15H	3.8	0.9337
16x16A	4.6	0.9287
17x17A	4.0	0.9368
17x17B	4.0	0.9355
17x17C	4.0	0.9349

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

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

Table 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}
6x6A	2.7††	0.7888†††
6x6B‡	2.7††	0.7824†††
6x6C	2.7††	0.8021†††
7x7A	2.7††	0.7974†††
7x7B	4.2	0.9386
8x8A	2.7††	0.7697†††
8x8B	4.2	0.9416
8x8C	4.2	0.9425
8x8D	4.2	0.9403
8x8E	4.2	0.9312
8x8F	4.0	0.9459

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

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

†† This calculation was performed for 3.0% planar-average enrichment, however, the authorized contents are limited to maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} value is conservative.

††† This calculation was performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68 is 0.0372 g/cm^2. Therefore, the listed maximum k_{eff} value is conservative.

‡ Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the specification of authorized contents, Chapter 1.

Table 6.1.2 (continued)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}
9x9A	4.2	0.9417
9x9B	4.2	0.9436
9x9C	4.2	0.9395
9x9D	4.2	0.9394
9x9E	4.0	0.9486
9x9F	4.0	0.9486
9x9G	4.2	0.9383
10x10A	4.2	0.9457††
10x10B	4.2	0.9436
10x10C	4.2	0.9433
10x10D	4.0	0.9376
10x10E	4.0	0.9185

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

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

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

Table 6.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}
6x6A	2.7††	0.7888
6x6B†††	2.7	0.7824
6x6C	2.7	0.8021
7x7A	2.7	0.7974
8x8A	2.7	0.7697

Note:

1. These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

2. These calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

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

†† These calculations were performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the specification of authorized contents, Chapter 1.

Table 6.1.4
SUMMARY OF THE CRITICALITY RESULTS FOR THE MOST REACTIVE ASSEMBLY FROM THE ASSEMBLY CLASSES IN EACH MPC
TO DEMONSTRATE COMPLIANCE WITH 10CFR71.55 AND 10CFR71.59

MPC-24, Assembly Class 15x15F, 4.1 wt% ^{235}U				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum‡ k_{eff}
Single Package, unreflected	100%	0%	n/a	0.9410
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	0.9397
Containment, fully reflected	100%	100%		0.9397
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	0.9436
Infinite Array of Undamaged Packages	0%	0%	10CFR71.59 (a)(1)	0.3950
MPC-68, Assembly Class 9x9E/F, 4.0 wt% ^{235}U				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum k_{eff}
Single Package, unreflected	100%	0%	n/a	0.9486
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	0.9470
Containment, fully reflected	100%	100%		0.9461
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	0.9468
Infinite Array of Undamaged Packages	0%	0%	10CFR71.59 (a)(1)	0.3808
MPC-68F, Assembly Class 6x6C, 2.7 wt% ^{235}U				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum k_{eff}
Single Package, unreflected	100%	0%	n/a	0.8021
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	0.8033
Containment, fully reflected	100%	100%		0.8033
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	0.8026
Infinite Array of Undamaged	0%	0%	10CFR71.59 (a)(1)	0.3034

‡ The maximum k_{eff} is equal to the sum of the calculated k_{eff}, two standard deviations, the code bias, and the uncertainty in the code bias. For cases with 100% internal moderation, the standard deviation is between 0.0007 and 0.0009, for cases with 0% internal moderation, the standard deviation is between 0.0002 and 0.0004.

Packages				

Table 6.1.4 (continued)
SUMMARY OF THE CRITICALITY RESULTS FOR THE MOST REACTIVE ASSEMBLY FROM THE ASSEMBLY CLASSES IN EACH MPC
TO DEMONSTRATE COMPLIANCE WITH 10CFR71.55 AND 10CFR71.59

MPC-24E/EF, Assembly Class 15x15F, 4.5 wt% ^{235}U				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum‡ k_{eff}
Single Package, unreflected	100%	0%	n/a	0.9495
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	0.9485
Containment, fully reflected	100%	100%		0.9486
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	0.9495
Infinite Array of Undamaged Packages	0%	0%	10CFR71.59 (a)(1)	0.4026
MPC-24E/EF TROJAN, Trojan Intact and Damaged Fuel, 3.7 wt% ^{235}U				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum k_{eff}
Single Package, unreflected	100%	0%	n/a	0.9377
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	0.9366
Containment, fully reflected	100%	100%		0.9377
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	0.9383
Infinite Array of Undamaged Packages	0%	0%	10CFR71.59 (a)(1)	0.3518
MPC-32, Assembly Class 15x15F *and 17x17C*~~, 4.0 wt% ^{235}U~~				
Configuration	% Internal Moderation	% External Moderation	Applicable Requirement	Maximum k_{eff}
Single Package, unreflected	100%	0%	n/a	*0.9480*~~79~~
Single Package, fully reflected	100%	100%	10CFR71.55 (b), (d), and (e)	*0.9469*~~78~~
Containment, fully reflected	100%	100%		*0.94854*
Infinite Array of Damaged Packages	100%	100%	10CFR71.59 (a)(2)	*0.9475*~~85~~
Infinite Array of Undamaged Packages	0%	0%	10CFR71.59 (a)(1)	*0.4248*~~111~~

‡ The maximum k_{eff} is equal to the sum of the calculated k_{eff}, two standard deviations, the code bias, and the uncertainty in the code bias. For cases with 100% internal moderation, the standard deviation is between 0.000~~7~~*4* and 0.0009, for cases with 0% internal moderation, the standard deviation is between 0.0002 and 0.0004.

Table 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E/EF

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**
14x14A	5.0	0.9380
14x14B	5.0	0.9312
14x14C	5.0	0.9365
14x14D	5.0	0.8875
14x14E	5.0	0.7651
15x15A	4.5	0.9336
15x15B	4.5	0.9487
15x15C	4.5	0.9462
15x15D	4.5	0.9445
15x15E	4.5	0.9471
15x15F	4.5	0.9495
15x15G	4.5	0.9062
15x15H	4.2	0.9455
16x16A	5.0	0.9358
17x17A	4.4	0.9447
17x17B	4.4	0.9438
17x17C	4.4	0.9433

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

Table 6.1.6

BOUNDING MAXIMUM k_{eff} VALUES IN THE MPC-24E/EF TROJAN

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Content	Maximum† k_{eff}
17x17B	3.7	Intact Fuel	0.9187
17x17B	3.7	Intact Fuel, Damaged Fuel and Fuel Debris	0.9377

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

Table 6.1.7

BOUNDING MAXIMUM k_{eff} VALUES IN THE MPC-32
FOR ASSEMBLIES NOT EXPOSED TO CONTROL RODS DURING IRRADIATION‡

Fuel Assembly Class	Maximum Allowable Enrichment†† (wt% ^{235}U)	Minimum Required Assembly Average Burnup†† (GWd/MTU)	Maximum† k_{eff}
15x15D, E, F, H	*1.8*	*0*	*0.92891*
	2.0	*10.65~~7.033~~*	*0.9472~~63~~*
	3.0	*28.63~~6.799~~*	*0.9466~~57~~*
	4.0	*40.43~~38.038~~*	*0.9444~~76~~*
	5.0	*52.94~~48.475~~*	*0.9480~~71~~*
17x17A, B, C	*1.8*	*0*	*0.9226~~5~~*
	2.0	*8.54~~5.568~~*	*0.9463~~58~~*
	3.0	*26.57~~5.000~~*	*0.9451~~71~~*
	4.0	*40.42~~38.712~~*	*0.9461~~8~~*
	5.0	*54.53~~1.449~~*	*0.9476~~2~~*

‡ *See Appendix 6.E for results for other conditions.*

†† Other combinations of maximum enrichment and minimum burnup have been evaluated which result in the same maximum k_{eff}. See Appendix 6.E for a bounding polynomial function.

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

6.3 MODEL SPECIFICATION

In compliance with the requirements of 10CFR71.31(a)(1), 10CFR71.33(a)(5), and 10CFR71.33(b), this section provides a description of the HI-STAR 100 System in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package.

6.3.1 Description of Calculational Model

Figures 6.3.1 through 6.3.3 show representative horizontal cross sections of the four types of cells used in the calculations, and Figures 6.3.4 through 6.3.6 illustrate the basket configurations used. Four different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket,
- an optimized 24 PWR assembly basket (MPC-24E/EF and Trojan MPC-24E/EF),
- a 32 PWR assembly basket, and
- a 68 BWR assembly basket.

For all basket designs, the same techniques and the same level of detail are used in the calculational models.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7, and conservatively neglecting the absorption in the overpack neutron shielding material (Holtite-A). Although the Boral neutron absorber panels are 156 inches in length, which is much longer than the active fuel length (maximum of 150 inches), they are assumed equal to the active fuel length in the calculations, except for the Trojan MPC-24E/EF. Due to the reduced height of the Trojan MPCs, there is the potential of a misalignment of about 1 inch between the active length and the Boral at the bottom of the active region. Conservatively, a misalignment of 3 inches is assumed in the calculational model for the Trojan MPCs. As shown on the drawings in Section 1.4, 16 of the 24 periphery Boral panels on the MPC-24 have reduced width (i.e., 6.25 inches wide as opposed to 7.5 inches). However, the calculational models for the MPC-24 conservatively assume all of the periphery Boral panels are 6.25 inches in width.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and Boral absorber panels on the stainless steel walls of the basket cells. Under normal conditions of transport, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ($k_{eff} < 0.5$). For the flooded condition (loading, unloading, and hypothetical accident condition), water was assumed to be present in the fuel rod pellet-to-clad gap regions (see Subsection 6.4.2.3 for justification). Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket designs in the HI-STAR 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included.

As indicated in Figures 6.3.1 through 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO and MCNP4a were used to determine the direction of the manufacturing tolerances which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity.

CASMO-3 and -4 were used for one of each of the two principal basket designs, i.e. for the fluxtrap design MPC-24 and for the non-fluxtrap design MPC-68. The effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. The conclusions in Table 6.3.1 are directly applicable to the MPC-24E/EF and the MPC-32, due to the similarity in the basket designs.

Additionaly, MCNP4a calculations are performed to evaluate the tolerances of the various basket dimensions of the MPC-68, MPC-24 and MPC-32 in further detail. The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a, with fuel assemblies centered in the fuel storage locations. Calculated k_{eff} results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR fuel assemblies. Each of the basket dimensions are evaluated for their minimum, nominal and maximum values. Due to the close similarity between the MPC-24 and MPC-24E, the basket dimensions are only evaluated for the MPC-24, and the same dimensional assumptions are applied to both MPC designs.

Based on the MCNP4a and CASMO calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined for the MPC basket designs. Because the reactivity effect (positive or negative) of the manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

The design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, and the fixed neutron absorbing panels (Boral). None of these parameters are affected by the hypothetical accident conditions of transport.

During the hypothetical accident conditions of transport, the HI-STAR 100 System is assumed to be flooded to such an extent as to cause the maximum reactivity and to have full water reflection to such an extent as to cause the maximum reactivity. Further, arrays of packages under the hypothetical accident conditions must be evaluated to determine the maximum number of packages that may be transported in a single shipment. Thus, the only differences between the normal and hypothetical accident condition calculational models are the internal/external moderator densities and the boundary conditions (to simulate an infinite array of HI-STAR 100 Systems).

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STAR 100 Systems are listed in Table 6.3.4. In this table, only the composition of fresh fuel is listed. For a discussion on the composition of spent fuel for burnup credit in the MPC-32 see Appendix 6.E.

The HI-STAR 100 System is designed such that the fixed neutron absorber (Boral) will remain effective for a period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of Boral are provided in Subsection 1.2.1.4.1.

The continued efficacy of the Boral is assured by acceptance testing, documented in Subsection 8.1.5.3, to validate the ^{10}B (poison) concentration in the Boral. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material, an MCNP4a calculation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of ^{10}B atoms destroyed is only 2.6E-09 in 50 years. Thus, the reduction in ^{10}B concentration in the Boral by neutron absorption is negligible. In addition, the structural analysis demonstrates that the sheathing, which affixes the Boral panel, remains in place during all hypothetical accident conditions, and thus, the Boral panel remains permanently fixed. Therefore, there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 9 of this SAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel positioning was judged to be not significant. Starting in Revision 10 of this SAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for all new or changed MPC designs or assembly classes. The calculations in this subsection serve to determine the highest maximum k_{eff} value for each of these assembly class and basket combinations, that is then reported in the summary tables in Section 6.1 and the results tables in Section 6.4. Further, the calculations in this subsection are used to determine the assembly class in each basket with the highest maximum k_{eff} that is then used to demonstrate compliance with the requirements of 10CFR71.55 and 10CFR71.59. All other calculations throughout this chapter, such as studies to determine bounding fuel dimension, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.

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

- Cell Center Configuration: All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;
- Basket Center Configuration: All assemblies in the basket are moved as closely 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 closely to the periphery of the basket as possible.

It needs to be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be $(1/5)^{24}$ or approximately 10^{-17} for the MPC-24, $(1/5)^{32}$ or approximately 10^{-23} for the MPC-32, and $(1/5)^{68}$ or approximately 10^{-48} for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

The results are presented in Table 6.3.5 for the MPC-24, Table 6.3.6 for the MPC-24E/EF, Table 6.3.7 for the Trojan MPC-24E/EF, and Table 6.3.8 for the MPC-68. For evaluations of eccentric fuel positions in the MPC-32 with burnup credit see Appendix 6.E. Each table shows the maximum k_{eff} value for centered and the two eccentric configurations for each of the assembly classes, and indicates the bounding configuration. The results are summarized as follows:

- In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.
- Most cases show the maximum reactivity for the basket center configuration, however, in some cases the reactivity is higher for the cell center configuration.

For each of the assembly class and basket combinations listed in Tables 6.3.5 through Table 6.3.8, the configuration showing the highest reactivity is used as the bounding configuration, and listed in the respective tables in Section 6.1. and 6.4. For evaluations of eccentric fuel positions in the MPC-32 with burnup credit see Appendix 6.E.

Table 6.3.1

CASMO-4 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter†	Δk for Maximum Tolerance		Action/Modeling Assumption
	MPC-24	MPC-68‡	
Reduce Boral Width to Minimum	N/A††† min. = nom. = 7.5" and 6.25"	N/A††† min. = nom. = 4.75"	Assume minimum Boral width
Increase UO_2 Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO_2 density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min.= 8.86" nom. = 8.92"	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" nom. = 8.92"	-0.0030 max. = 6.113" nom. = 6.053"	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" nom. = 1.15"	N/A	Assume minimum water gap in the MPC-24

† Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

‡ Calculations for the MPC-68 were performed with CASMO-3 [6.3.1 – 6.3.4].

††† The Boral width for the MPC-68 is 4.75" +0.125", -0" , the Boral widths for the MPC-24 are 7.5" +0.125", -0" and 6.25" +0.125" –0" (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

CASMO-4 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	Δk Maximum Tolerance		Action/Modeling Assumption
	MPC-24	MPC-68‡	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0030	-0.0039	
70°C	-0.0089	-0.0136	
100°C	-0.0162	-0.0193	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0251	-0.0241	
100°C	-0.0412	-0.0432	
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

‡ Calculations for the MPC-68 were performed with CASMO-3 [6.3.1 – 6.3.4].

Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated k_{eff}
MPC-24†† (17x17A01 @ 4.0% Enrichment)						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9325±0.0008†††
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9300±0.0008
nominal	(10.906")	nom. –0.04"	(8.88")	nom. + 0.05"	(0.3625")	0.9305±0.0007
MPC-68 (8x8C04 @ 4.2% Enrichment)						
minimum	(6.43")	minimum	(5.993")	nominal	(1/4")	0.9307±0.0007
nominal	(6.49")	nominal	(6.053")	nominal	(1/4")	0.9274±0.0007
maximum	(6.55")	maximum	(6.113")	nominal	(1/4")	0.9272±0.0008
nom. + 0.05"	(6.54")	nominal	(6.053")	nom. + 0.05"	(0.30")	0.9267±0.0007

Note: Values in parentheses are the actual value used.

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

†† All calculations for the MPC-24 assume minimum water gap thickness (1.09").

††† Numbers are 1σ statistical uncertainties.

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-32 (17x17A @ 4.0% Enrichment)			
minimum (9.158")	minimum (8.73")	nominal (9/32")	*0.9105~~236~~±0.0005*†††
nominal (9.218")	nominal (8.79")	nominal (9/32")	*0.9098~~200~~±0.00054*
maximum (9.278")	maximum (8.85")	nominal (9/32")	*0.9082~~204~~±0.00054*
nominal+0.05" (9.268")	nominal (8.79")	nominal+0.05" (0.331")	*0.9089~~199~~±0.0005*
minimum+0.05"(9.208")	minimum (8.73")	nominal+0.05" (0.331")	*0.9104~~204~~±0.00054*
maximum (9.278")	Maximum-0.05" (8.80")	nominal+0.05" (0.331")	*0.9090~~196~~±0.0005*

Notes:

1. Values in parentheses are the actual value used.

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

††† *Numbers are 1σ statistical uncertainties.*

Table 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Pitch	Box I.D.	Box Wall Thickness	Water-Gap Flux Trap
MPC-24	nominal (10.906")	maximum (8.98")	nominal (5/16")	minimum (1.09")
MPC-24E	nominal (10.847")	maximum (8.81", 9.11" for DFC Positions, 9.36" for DFC Positions in Trojan MPC)	nominal (5/16")	minimum (1.076", 0.776" for DFC Positions, 0.526" for DFC Positions in Trojan MPC)
MPC-32	minimum (9.158")	minimum (8.73")	nominal (9/32")	N/A
MPC-68	minimum (6.43")	minimum (5.993")	nominal (1/4")	N/A

Table 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC-24		
UO_2 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
BORAL (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
MPC-32		
BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC-68		
UO_2 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	Atom-Density	Wgt. Fraction
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
UO_2 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	Atom-Density	Wgt. Fraction
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
MOX FUEL†, DENSITY (g/cc) = 10.522		
Nuclide	Atom-Density	Wgt. Fraction
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

† The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.798E-02	1.212E-01
92235	4.001E-04	1.484E-02
92238	2.742E-05	1.030E-03
90232	2.357E-02	8.630E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

COMMON MATERIALS		
ZR CLAD, DENSITY (g/cc) = 6.550		
Nuclide	**Atom-Density**	**Wgt. Fraction**
40000	4.323E-02	1.000E+00
MODERATOR (H_2O), DENSITY (g/cc) = 1.000		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01
STAINLESS STEEL, DENSITY (g/cc) = 7.840		
Nuclide	**Atom-Density**	**Wgt. Fraction**
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
ALUMINUM, DENSITY (g/cc) = 2.700		
Nuclide	**Atom-Density**	**Wgt. Fraction**
13027	6.026E-02	1.000E+00

Table 6.3.5

EFFECT OF ECCENTRIC FUEL POSITIONING IN THE MPC-24

Fuel Assembly Class	Maximum k_{eff}			Bounding Configuration	Bounding Maximum k_{eff}
	Cell Center Configuration	Basket Center Configuration	Basket Periphery Configuration		
14x14A	0.9296	0.9271	0.8951	Cell Center	0.9296
14x14B	0.9228	0.9207	0.8904	Cell Center	0.9228
14x14C	0.9287	0.9307	0.9068	Basket Center	0.9307
14x14D	0.8507	0.8498	0.8225	Cell Center	0.8507
14x14E	0.7627	0.7608	0.7003	Cell Center	0.7627
15x15A	0.9204	0.9227	0.9037	Basket Center	0.9227
15x15B	0.9388	0.9388	0.9240	Basket Center	0.9388
15x15C	0.9361	0.9351	0.9218	Cell Center	0.9361
15x15D	0.9367	0.9364	0.9248	Cell Center	0.9367
15x15E	0.9368	0.9392	0.9264	Basket Center	0.9392
15x15F	0.9395	0.9410	0.9271	Basket Center	0.9410
15x15G	0.8876	0.8907	0.8761	Basket Center	0.8907
15x15H	0.9337	0.9335	0.9214	Cell Center	0.9337
16x16A	0.9287	0.9284	0.9051	Cell Center	0.9287
17x17A	0.9368	0.9362	0.9221	Cell Center	0.9368
17x17B	0.9324	0.9355	0.9204	Basket Center	0.9355
17x17C	0.9336	0.9349	0.9225	Basket Center	0.9349

Table 6.3.6

EFFECT OF ECCENTRIC FUEL POSITIONING IN THE MPC-24E/EF

Fuel Assembly Class	Maximum k_{eff}			Bounding Configuration	Bounding Maximum k_{eff}
	Cell Center Configuration	Basket Center Configuration	Basket Periphery Configuration		
14x14A	0.9380	0.9327	0.9080	Cell Center	0.9380
14x14B	0.9312	0.9288	0.9029	Cell Center	0.9312
14x14C	0.9356	0.9365	0.9189	Basket Center	0.9365
14x14D	0.8875	0.8857	0.8621	Cell Center	0.8875
14x14E	0.7651	0.7536	0.7001	Cell Center	0.7651
15x15A	0.9336	0.9304	0.9188	Cell Center	0.9336
15x15B	0.9465	0.9487	0.9367	Basket Center	0.9487
15x15C	0.9462	0.9452	0.9348	Cell Center	0.9462
15x15D	0.9440	0.9445	0.9343	Basket Center	0.9445
15x15E	0.9455	0.9471	0.9372	Basket Center	0.9471
15x15F	0.9468	0.9495	0.9406	Basket Center	0.9495
15x15G	0.9054	0.9062	0.8970	Basket Center	0.9062
15x15H	0.9423	0.9455	0.9365	Basket Center	0.9455
16x16A	0.9341	0.9358	0.9183	Basket Center	0.9358
17x17A	0.9447	0.9443	0.9355	Cell Center	0.9447
17x17B	0.9421	0.9438	0.9303	Basket Center	0.9438
17x17C	0.9433	0.9431	0.9347	Cell Center	0.9433

Table 6.3.7

EFFECT OF ECCENTRIC FUEL POSITIONING IN THE TROJAN MPC-24E/EF

Fuel Assembly Class	Maximum k_{eff}			Bounding Configuration	Bounding Maximum k_{eff}
	Cell Center Configuration	Basket Center Configuration	Basket Periphery Configuration		
17x17B (Intact Fuel)	0.9161	0.9187	0.9059	Basket Center	0.9187
17x17B (Intact Fuel and Damaged Fuel/Fuel Debris)	0.9377	0.9353	0.9338	Cell Center	0.9377

Table 6.3.8

EFFECT OF ECCENTRIC FUEL POSITIONING IN THE MPC-68

Fuel Assembly Class	Maximum k_{eff}			Bounding Configuration	Bounding Maximum k_{eff}
	Cell Center Configuration	Basket Center Configuration	Basket Periphery Configuration		
8x8F	0.9411	0.9459	0.9193	Basket Center	0.9459
9x9E/F	0.9401	0.9486	0.9166	Basket Center	0.9486
9x9G	0.9309	0.9383	0.9124	Basket Center	0.9383

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational or Experimental Method

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V†, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, calculations assuming fresh fuel used a minimum of 5,000 simulated histories 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). For parameters used in the burnup credit calculations see Appendix 6.E. Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket in the HI-STAR 100 System.

CASMO-4 [6.1.10-6.1.12] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO calculations. Additionally, CASMO-4 was used to determine the isotopic composition of spent fuel for burnup credit in the MPC-32 (see Appendix 6.E).

† For burnup credit calculations in the MPC-32, ENDF/B-VI cross sections are used for nuclides where ENDF/B-V cross sections are not available.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate the candidate fuel assemblies with enrichments indicated in Tables 6.1.1 through 6.1.3 and 6.1.5 through 6.1.7. The calculations were based on the assumption that the HI-STAR 100 System was fully flooded with water. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

6.4.2.1 Internal and External Moderation

The regulations in 10CFR71.55 include the requirement that the system remains subcritical when assuming moderation to the most reactive credible extent. The regulations in 10CFR71.59 require subcriticality for package arrays under different moderation conditions. The calculations in this section demonstrate that the HI-STAR 100 System remains subcritical for all credible conditions of moderation, and that the system fulfills all requirements of 10CFR71.55 and 10CFR71.59. The following subsections 6.4.2.1.1 through 6.4.2.4 present various studies to confirm or identify the most reactive configuration or moderation condition. Specifically, the following conditions are analyzed:

- Reduced internal and external water density for single packages (6.4.2.1.1) and package arrays (6.4.2.1.2);
- Variation in package to package distance in package arrays (6.4.2.1.2);
- Partial internal flooding of package (6.4.2.2);
- Flooding of pellet to cladding gap of the fuel rods (6.4.2.3); and
- Preferential flooding, i.e. uneven flooding inside the package (6.4.2.4).

The calculations that specifically demonstrate compliance with the individual requirements of 10CFR71.55 and 10CFR71.59 are presented in Section 6.4.3. These calculations are performed for all MPCs.

The studies in subsections 6.4.2.1.1 through 6.4.2.4 have been performed for both principal basket designs (flux-trap and non-flux-trap) and for both fuel designs (BWR and PWR). Specifically, the studies are performed with the MPC-24 (flux-trap design / PWR fuel) and the MPC-68 (non-flux-trap design / BWR fuel). The results of the studies show a consistent behavior of the different basket designs and fuel types for different moderation conditions. Consequently, the conclusions drawn from these studies are directly applicable to the remaining baskets, namely the MPC-24E/EF (flux-trap design, PWR), MPC-32 (non-flux-trap design, PWR) and MPC-68F (non-flux-trap design, BWR), and no further studies are required for these baskets.

The studies in subsection 6.4.2.1.1 through 6.4.2.4 have been performed with the fuel assemblies centered in each storage location in the basket, which is not necessarily the most reactive position. However, this assumption is acceptable since the objective of these studies is to determine the most reactive moderation condition, not the highest reactivity. The calculations in Section 6.4.3 that demonstrate compliance with 10CFR71.55 and 19CFR71.59 are performed with the most reactive assembly position as discussed in Section 6.3.3.

Regarding the effect of low moderator density it is noted that with a neutron absorber present (i.e., the Boral sheets on the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur when strong neutron absorbing material is present or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask and for infinite arrays of casks were made to confirm that the phenomenon does not occur with low density water inside or outside the HI-STAR 100 Systems.

6.4.2.1.1 Single Package Evaluation

Calculations for a single package are performed for the MPC-24 and MPC-68. The Calculational model consists of the HI-STAR System surrounded by a rectangular box filled with water. The neutron absorber on the outside of the HI-STAR is neglected, since it might be damaged under accident conditions, and since it is conservative to replace the neutron absorber (Holtite-A) with a neutron reflector (water). The minimum water thickness on each side of the cask is 30 cm, which effectively represents full water reflection. The outer surfaces of the surrounding box are conservatively set to be fully reflective, which effectively models a three dimensional array of cask systems with an minimum surface to surface distance of 60 cm. The calculations with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case 1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Figure 6.4.9 plots calculated k_{eff} values ($\pm 2\sigma$) as a function of internal moderator density for both MPC designs with 100% external moderator density (i.e., full water reflection).

Results listed in Table 6.4.1 and plotted in Figure 6.4.9 support the following conclusions:

- The calculated k_{eff} for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and

- Reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STAR 100 System.

6.4.2.1.2 Evaluation of Package Arrays

In terms of reactivity, the normal conditions of transport (i.e., no internal or external moderation) are bounded by the hypothetical accident conditions of transport. Therefore, the calculations in this section evaluate arrays of HI-STAR 100 Systems under hypothetical accident conditions (i.e, internal and external moderation by water to the most reactive credible extent and no neutron shield present).

In accordance with 10CFR71.59 requirements, calculations were performed to simulate an infinite three-dimensional square array of internally fully-flooded (highest reactivity) casks with varying cask spacing and external moderation density. The MPC-24 was used for this analysis. The maximum k_{eff} results of these calculations are listed in Table 6.4.2 and confirm that the individual casks in a square-pitched array are independent of external moderation and cask spacing. The maximum value listed in Table 6.4.2 is statistically equivalent (within three standard deviations) to the reference value (Case 1 shown in Table 6.4.1) for a single unreflected fully flooded cask.

To further investigate the reactivity effects of array configurations, calculations were also performed to simulate an infinite three-dimensional hexagonal (triangular-pitched) array of internally fully-flooded (highest reactivity) MPC-24 casks with varying cask spacing and external moderation density. The maximum k_{eff} results of these calculations are listed in Table 6.4.3 and confirm that the individual casks in a hexagonal (triangular pitched) array are effectively independent of external moderation and cask spacing. The maximum value listed in Table 6.4.3 is statistically equivalent (within two standard deviations) to the reference value (Case 1 shown in Table 6.4.1) for a single unreflected fully flooded cask.

To assure that internal moderation does not result in increased reactivity, hexagonal array calculations were also performed for 10% internal moderator with 10% and 100% external moderation for varying cask spacing. Maximum k_{eff} results are summarized in Table 6.4.4 and confirm the very low values of k_{eff} for low values of internal moderation.

The results presented thus far indicate that neutronic interaction between casks is not enhanced by the neighboring casks or the water between the neighboring casks, and thus, the most reactive arrangement of casks corresponds to a tightly packed array with the cask surfaces touching. Therefore, calculations were performed for an infinite hexagonal (triangular pitched) array of touching casks (neglecting the Holtite-A neutron shield). These calculations were performed for the MPC-24 and the MPC-68 designs, in the internally flooded (highest reactivity) and internally dry conditions, with and without external flooding. The results of these calculations are listed in Table 6.4.5. For both the MPC-24 and MPC-68, the maximum k_{eff} values are shown to be statistically equivalent (within one standard deviation) to that of a single internally flooded unreflected cask and are below the regulatory limit of 0.95.

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

6.4.2.2 Partial Flooding

To demonstrate that the HI-STAR 100 System 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 the HI-STAR 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for the MPC-24 and MPC-68 designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). Results of these calculations are shown in Table 6.4.6. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. This conclusion is also true for the other baskets that were not analyzed under partial flooding conditions, since increasing the water level always improves the moderation condition of the fuel and therefore results in an increase in reactivity†. The fully flooded case therefore represents the bounding condition for all MPC basket types.

6.4.2.3 Clad Gap Flooding

The reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.7 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded.

6.4.2.4 Preferential Flooding

† The rate of increase in reactivity along the fuel length, though, could be different between different MPC designs. An example would be the MPC-32 with burnup credit where the reactivity is strongly affected by the lower burned ends of the fuel.

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

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits. Because the fuel cladding temperatures remain below their design limits (as demonstrated in Chapter 3) and the inertial loading remains below 63g's (Section 2.9), the cladding remains intact. For damaged BWR fuel assemblies and BWR fuel debris, the assemblies or debris are pre-loaded into stainless steel Damaged Fuel Containers fitted with 250x250 fine mesh screens (20x20 for Trojan FFC) which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked and the MPC fuel baskets cannot be preferentially flooded.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. This condition would be the result of the water tension across the mesh screens. The maximum water level inside the DFCs for this condition is calculated from the dimensions of the mesh screen and the surface tension of water. The wetted perimeter of the screen openings is up to 50 ft per square inch of screen. With a surface tension of water of 0.005 lbf/ft, this results in a maximum pressure across the screen of 0.25 psi, corresponding to a maximum water height in the DFC of 7 inches. For added conservativism, a value of 12 inches is used. Assuming this condition, calculations are performed for the two possible DFC configurations:

- MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A, see Subsection 6.4.4)
- MPC-24E or MPC-24EF with 4 DFCs and 20 intact assemblies (Bounding all PWR assembly classes, see Subsection 6.4.9)

For each configuration, the case resulting in the highest maximum k_{eff} for the fully flooded condition (see Subsections 6.4.4 and 6.4.9) is re-analyzed assuming the preferential flooding condition. For these analyses, the lower 12 inches of the active fuel in the DFCs and the water region below the active fuel (see Figure 6.3.7) are filled with full density water (1.0 g/cc). The remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). All calculations are performed for a single unreflected cask. Table 6.4.10 lists the maximum k_{eff} for the configurations in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{eff} than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

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 Subsections 6.4.4 and 6.4.9.

6.4.2.5 Hypothetical Accidents Conditions of Transport

The analyses presented in Section 2.7 of Chapter 2 and Section 3.5 of Chapter 3 demonstrate that the damage resulting from the hypothetical accident conditions of transport are limited to a loss of the neutron shield material as a result of the hypothetical fire accident. Because the criticality analyses do not take credit for the neutron shield material (Holtite-A), this condition has no effect on the criticality analyses.

As reported in Table 2.7.1, the minimum factor of safety for all MPCs as a result of the hypothetical accident conditions of transport is larger than 1.0 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change in the MPC-24 and MPC-24E/EF will be insignificant compared to the characteristic dimension of the flux trap.

Regarding the fuel assembly integrity, SAR Section 2.9 contains an evaluation of the fuel under accident conditions that concludes that the fuel rod cladding remains intact under the design basis deceleration levels set for the HI-STAR 100.

In summary, the hypothetical transport accidents have no adverse effect on the geometric form of the package contents important to criticality safety, and thus, are limited to the effects on internal and external moderation evaluated in Subsection 6.4.2.1.

6.4.3 Criticality Results

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

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

where:

- ⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;
- ⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final k_{eff} value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93.

However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;

⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code (MCNP4a or KENO5a);

⇒ *Bias* is the systematic error in the calculations (code dependent) determined by comparison with critical experiments ~~in Appendix 6.A~~; and

⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level~~; see Appendix 6.A~~).

Appendix 6.A presents the critical experiment benchmarking *for fresh UO_2 and MOX fuel* and the derivation of the *corresponding* bias and standard error of the bias (95% probability at the 95% confidence level).

See Appendix 6.E, Section 6.E.3, for the critical experiment benchmarking for spent fuel.

The studies in sections 6.4.2.1 through 6.4.2.4 demonstrate that the moderation by water to the most reactive credible extent corresponds to the internally fully flooded condition of the MPC, with the pellet-to-clad gap in the fuel rods also flooded with water. The external moderation and/or the presence of other surrounding packages, however, has a statistically negligible effect. To demonstrate compliance with 10CFR71.55 and 10CFR71.59, the following set of four calculations is performed for each of the MPC designs:

- Single containment with full internal and external water moderation. The full external water moderation is modeled through an infinite array of containments with a 60cm surface to surface distance. The containment system corresponds to the 2.5 inch inner shell of the overpack. This case addresses the requirement of 10CFR71.55 (b).
- Single cask with full internal and external water moderation. As for the single containment, the full external water moderation is modeled through an infinite array. The external neutron moderator is conservatively neglected in the model. This case also addresses the requirement of 10CFR71.55 (b).
- Hexagonal array of touching casks with full internal and external water reflection. This addresses the requirement of 10CFR71.59 (a)(2) and the determination of the transport index based on criticality control according to 10CFR71.59 (b).
- Hexagonal array of touching casks, internally and externally dry. This addresses the requirement of 10CFR71.59 (a) (1) and the determination of the transport index based on criticality control according to 10CFR71.59 (b). This also addresses the requirement of 10CFR71.55 (d)(1).

To satisfy the requirements of 10CFR71.55 (b)(1), the calculations are performed

- with the assembly type that results in the highest reactivity in the MPC. This is the assembly class 15x15F for the MPC-24, MPC-24E/EF and MPC-32, the assembly class 17x17B with intact and damaged assemblies in the Trojan MPC-24E/EF, the assembly class 9x9E/F in the MPC-68, and the assembly class 6x6C for the MPC-68F; and
- with the bounding basket dimensions as determined in Section 6.3.1 for each basket; and
- with eccentric fuel positioning as necessary, as discussed in Section 6.3.3.

The maximum k_{eff} values for all these cases, calculated with 95% probability at the 95% confidence level, are listed in Table 6.4.12. Results of the criticality safety calculations for other assembly classes under the condition of full internal flooding with water are summarized in Section 6.1. Corresponding detailed results including the maximum k_{eff}, standard deviation and energy of the average lethargy causing fission (EALF) are listed for all MPCs except the MPC-32 in Tables 6.4.13 through 6.4.17. Results for the MPC-32 are presented in Appendix 6.E. Overall, these results confirm that for each of the candidate fuel assemblies 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 and hypothetical accident conditions of transport. Therefore, compliance with 10CFR71.55 for single packages and 10CFR71.59 for package arrays in both normal and hypothetical accident conditions of transport is demonstrated for all of the fuel assembly classes and basket configurations listed in Tables 6.1.1 through 6.1.3 and 6.1.5 through 6.1.7. It further demonstrates that the transportation index for criticality control is zero because an infinite number of HI-STAR 100 casks will remain subcritical (k_{eff}<0.95) under both normal and hypothetical accident conditions of transport.

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.

Tables listing the maximum k_{eff}, calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies in each assembly class for the MPC-24, MPC-68 and MPC-68F basket configurations, and with assemblies centered in the fuel storage locations, are provided in Section 6.2.

6.4.4 Damaged Fuel Container for BWR Fuel

Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs). Two different DFC types with slightly different cross sections are analyzed. DFCs containing fuel debris must be stored in the MPC-68F. DFCs containing damaged fuel assemblies may be stored in either the MPC-68 or MPC-68F. Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 has a higher specified ^{10}B

loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68 Although the maximum planar-average enrichment of the damaged fuel is limited to 2.7% ^{235}U as specified in Chapter 1, analyses have been made for three possible scenarios, conservatively assuming fuel†† of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.1 through 6.4.7.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This configuration is illustrated in Figure 6.4.8.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.8, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in ^{238}U neutron capture (higher effective resonance integral for ^{238}U absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.8.

The analyses performed and summarized in Table 6.4.8 provides the relative magnitude of the effects on the reactivity. This information in combination with the maximum k_{eff} values listed in Table 6.1.3 and the conservatism in the analyses, demonstrate that the maximum k_{eff} of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of $k_{eff} < 0.95$.

Appendix 6.D provides sample input files for the damaged fuel analysis.

6.4.5 Fuel Assemblies with Missing Rods

†† 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.10. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in ^{235}U (equivalent to UO_2 fuel with an enrichment of approximately 1.7 wt% ^{235}U), the large spacing between the rods (the pitch is approximately 1", the cladding OD is 0.412") and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in Tables 6.1.2 and 6.1.3 this result demonstrates, that the k_{eff} for a Thoria Rod Canister loaded into the MPC68 or the MPC68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of $k_{eff} < 0.95$.

6.4.7 Sealed Rods Replacing BWR Water Rods

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

6.4.8 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STAR 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a keff less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

6.4.9 PWR Damaged Fuel and Fuel Debris

The MPC-24E, MPC-24EF, and Trojan MPC-24E and MPC-24EF are designed to contain damaged fuel and fuel debris, loaded into Damaged Fuel Containers (DFCs) or Failed Fuel Cans (FFCs). There is one generic DFC for the MPC-24E/EF, and two containers, a Holtec DFC and a Trojan FFC for the Trojan MPC-24E/EF. In this section, the term "DFC" is used to specify either of these components. In any case, the number of DFCs is limited to 4, and the permissible locations of the DFCs are shown in Figure 6.4.11.

Only the Trojan MPC-24E/EF is certified for damaged fuel and fuel debris. However, the generic MPC-24E/EF is also designed to accommodate damaged fuel debris, and the majority of criticality evaluations for damaged fuel and fuel debris are performed for the generic MPC-24E/EF, with only a smaller number of calculations performed for the Trojan MPCs. Therefore, criticality evaluations for both the generic MPC-24E/EF and the Trojan MPC-24E/EF are presented in this subsection, even though the Trojan MPC-24E/EF are the only MPCs authorized to transport damaged fuel and fuel debris.

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 Chapter 1). Therefore, apart from possible missing fuel rods, damaged fuel assemblies have the same geometric configuration as intact fuel assemblies and consequently the same reactivity. Missing fuel rods can result in a slight increase of reactivity. After a drop accident, however, it can not be assumed that the initial geometric integrity is still maintained. For a drop on either the top or bottom of the cask, the damaged fuel assemblies could collapse. This would result in a configuration with a reduced length, but increased amount of fuel per unit length. For a side drop, fuel rods could be compacted to one side of the DFC. In either case, a significant relocation of fuel within the DFC is possible, which creates a greater amount of fuel in some areas of the DFC, whereas the amount of fuel in other areas is reduced. Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

In the cases of fuel debris or relocated damaged fuel, there is the potential that fuel could be present in axial sections of the DFCs that are outside the basket height covered with Boral. However, in these sections, the DFCs are not surrounded by any intact fuel, only by basket cell walls, non-fuel hardware and water. Studies have shown that this condition does not result in any significant effect on reactivity, compared to a condition where the damaged fuel and fuel debris is restricted to the axial section of the basket covered by Boral. All calculations for damaged fuel and fuel debris are therefore performed assuming that fuel is present only in the axial sections covered by Boral, and the results are directly applicable to any situation where damaged fuel and fuel debris is located outside these sections in the DFCs.

To address all the situations listed above and identify the configuration or configurations leading to the highest reactivity, it is impractical to analyze a large number of different geometrical configurations for each of the fuel classes. Instead, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following sections.

All calculations for generic damaged fuel and fuel debris are performed using a full cask model with the maximum permissible number of Damaged Fuel Containers. For the MPC-24E and MPC-24EF, the model consists of 20 intact assemblies, and 4 DFCs in the locations shown in Figure 6.4.11. The bounding assumptions regarding the intact assemblies and the modeling of the damaged fuel and fuel debris in the DFCs are discussed in the following sections.

6.4.9.1 Bounding Intact Assemblies

Intact PWR assemblies stored together with DFCs in the MPC-24E/EF are limited to a maximum enrichment of 4.0 wt% ^{235}U, regardless of the fuel class. Results presented in Table 6.1.5 for the MPC-24E/EF loaded with intact assemblies only are for different enrichments for each class, ranging between 4.2 and 5.0 wt% ^{235}U, making it difficult to directly identify the bounding assembly. However, the assembly class 15x15H is among the classes with the highest reactivity, but has the lowest initial enrichment. Therefore, the 15x15H assembly is used as the intact PWR assembly for all calculations with DFCs.

The Trojan MPC-24E/EF is only certified for the assembly class 17x17B, which bounds the fuel types used at the Trojan plant. Consequently, the assembly class 17x17B is used as the intact assembly in all calculations for the Trojan MPC-24E/EF.

6.4.9.2 Bare Fuel Rod Arrays

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

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

- The active length of these rods is chosen to be the maximum active fuel length of all fuel assemblies listed in Section 6.2, which is 150 inch for PWR fuel.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 64 (8x8) and 729 (27x27) for PWR fuel.

- Analyses are performed for the minimum, maximum and typical pellet diameter of the fuel.

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

This is also a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 150 inch. For some of the analyzed cases, this assumption results in more uranium mass being modeled in the DFCs than is permitted by the uranium mass loading restrictions listed in Chapter 1.

To demonstrate the level of conservatism, additional analyses are performed with the DFC containing various realistic assembly configurations such as intact assemblies, assemblies with missing fuel rods and collapsed assemblies, i.e. assemblies with increased number of rods and decreased rod pitch.

As discussed in Subsection 6.4.9, all calculations are performed for full cask models, containing the maximum permissible number of DFCs together with intact assemblies.

Graphical presentations of the calculated maximum k_{eff} for each case as a function of the fuel mass per unit length of the DFC are shown in Figure 6.4.12. The results for the bare fuel rods show a distinct peak in the maximum k_{eff} at about 3.5 $kgUO_2$/inch.

The realistic assembly configurations are typically about 0.01 (delta-k) or more below the peak results for the bare fuel rods, demonstrating the conservatism of this approach to model damaged fuel and fuel debris configurations such as severely damaged assemblies and bundles of fuel rods.

For fuel debris configurations consisting of bare fuel pellets only, the fuel mass per unit length would be beyond the value corresponding to the peak reactivity. For example, for DFCs filled with a mixture of 60 vol% fuel and 40 vol% water the fuel mass per unit length is 7.92 $kgUO_2$/inch for the PWR DFC. The corresponding reactivities are significantly below the peak reactivies. The difference is about 0.01 (delta-k) or more for PWR fuel. Furthermore, the filling height of the DFC would be less than 70 inches in these examples due to the limitation of the fuel mass per basket position, whereas the calculation is conservatively performed for a height of 150 inch. These results demonstrate that even for the fuel debris configuration of bare fuel pellets, the model using bare fuel rods is a conservative approach.

To demonstrate that the bare fuel rod approach also bounds the potential presence of fuel fragments in the DFCs, additional calculations were performed with fuel fragments in the DFCs instead of bare fuel rods. The fuel fragments are modeled as regular 3-dimensional arrays of fuel cubes positioned inside water cubes. Both the dimension of the fuel cubes and the fuel-to-water-volume ratio are varied over a wide range. Calculations are performed for the MPC-24E/EF Trojan, and the results are presented in Table 6.4.18. The highest maximum k_{eff} is 0.9320 for a fragment outer dimension of 0.2 inches and a fuel to water volume ratio of 0.4. This maximum k_{eff} value is lower than the corresponding value for the bare fuel rod model, which is 0.9377 as shown in Table 6.4.17. The damaged fuel and fuel debris model based on bare fuel rods therefore bounds any condition involving fuel fragments in the DFCs.

6.4.9.3 Results for MPC-24E and MPC-24EF

The MPC-24E is designed for the storage of up to four DFCs with damaged fuel in the four outer fuel baskets cells shaded in Figure 6.4.11. The MPC-24EF allows storage of up to four DFCs with damaged fuel or fuel debris in these locations. These locations are designed with a larger box ID to accommodate the DFCs. For an enrichment of 4.0 wt% ^{235}U for the intact fuel, damaged fuel and fuel debris, the results for the various configurations outlined in Subsection 6.4.9.2 are summarized in Figure 6.4.12 and in Table 6.4.11. Figure 6.4.12 shows the maximum k_{eff}, including bias and calculational uncertainties, for various actual and hypothetical damaged fuel and fuel debris configurations as a function of the fuel mass per unit length of the DFC. For the intact assemblies, the 15x15H assembly class was chosen (see Subsection 6.4.9.1). Table 6.4.11 lists the highest maximum k_{eff} for the various configurations. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.4.9.4 Results for Trojan MPC-24E and MPC-24EF

For the Trojan MPC-24E/EF, bare fuel rod arrays with arrays sizes between 11x11 and 23x23 were analyzed as damaged fuel/fuel debris, with a pellet diameter corresponding to the 17x17B assembly class. The highest maximum k_{eff} value is shown in Table 6.1.6, and is below the 0.95 regulatory limit. The realistic damaged fuel assembly configurations in the DFC, such as assemblies with missing rods, were not analyzed in the Trojan MPC-24E/EF since the evaluations for the generic MPC-24E/EF demonstrate that these conditions are bounded by the fuel debris model using bare fuel pellets.

6.4.10 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware such as Thimble Plugs (TPs), Burnable Poison Rod Assemblies (BPRAs), Rod Cluster Control Assemblies (RCCAs) and similar devices are permitted for storage with the PWR fuel assemblies in the Trojan MC-24E/EF. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged.

Therefore, 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.

6.4.11 Reactivity Effect of Potential Boral Damage

During the manufacturing process of the fuel baskets, it is possible that minor damage to Boral panels occurs during welding operations. Criticality calculations have been performed for all basket types to determine whether this condition could have an effect on the reactivity of the system. Since the potential Boral damage is typically the result of welding operations, the damage would occur in a narrow area along the edge of the panel, and would only be present in a few panels within each basket. However, in order to maximize the potential reactivity effect of the damage in the calculations, it is assumed that the damage occurs in an area with a diameter of 1 inch at the center of the Boral panel, and that this condition exists in every panel in the basket. It is further assumed that the Boral in this area is completely replaced by water, while in reality only a relocation of the Boral would occur, since the Boral is completely covered by the sheathing. Calculations performed under these assumption demonstrate that the conservatively modeled Boral damage has a negligible effect on the reactivity, i.e. the difference to the condition without the damage is less than 2 standard deviations. For example, for the MPC-24 and MPC-24E, the change in reactivity is +0.0006 and -0.0004, respectively, for a standard deviation between 0.0004 and 0.0005. In the MPC-24E for Trojan, a specific potential damage was identified that is not bounded by the generic approach described above. To demonstrate that this condition is acceptable, a specific calculation was performed assuming a damage of 5 square-inches in a specific location in up to 8 Boral panels in the basket, and was found to have again a negligible effect on reactivity. In summary, these calculations demonstrate that Boral damage bounded by the configurations assumed in the analyses is acceptable and does not affect the reactivity of the HI-STAR System.

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS†
WITH MPC-24 AND MPC-68

	Water Density		MCNP4a Results					
Case Number	Internal	External	MPC-24 (17x17A01 @ 4.0%)			MPC-68 (8x8C04 @ 4.2%)		
			Max. k_{eff}††	1 σ	EALF (eV)	Max. k_{eff}	1 σ	EALF (eV)
1	100%	single cask	0.9368	0.0008	0.2131	0.9348	0.0007	0.2915
2	100%	100%	0.9354	0.0009	0.2136	0.9339	0.0005	0.2922
3	100%	70%	0.9362	0.0008	0.2139	0.9339	0.0006	0.2921
4	100%	50%	0.9352	0.0008	0.2144	0.9347	0.0004	0.2924
5	100%	20%	0.9372	0.0008	0.2138	0.9338	0.0005	0.2921
6	100%	10%	0.9380	0.0009	0.2140	0.9336	0.0005	0.2920
7	100%	5%	0.9351	0.0008	0.2142	0.9333	0.0006	0.2936
8	100%	0%	0.9342	0.0008	0.2136	0.9338	0.0005	0.2922
9	70%	0%	0.8337	0.0007	0.4115	0.8488	0.0004	0.6064
10	50%	0%	0.7426	0.0008	0.8958	0.7631	0.0004	1.4515
11	20%	0%	0.5606	0.0007	15.444	0.5797	0.0006	26.5
12	10%	0%	0.4834	0.0005	160.28	0.5139	0.0003	241
13	5%	0%	0.4432	0.0004	1133.9	0.4763	0.0003	1770
14	10%	100%	0.4793	0.0005	171.79	0.4946	0.0003	342

† For an infinite square array of casks with 60 cm spacing between cask surfaces.

†† Maximum k_{eff} includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

REACTIVITY EFFECTS OF SPACING AND WATER MODERATOR DENSITY FOR SQUARE ARRAYS OF MPC-24 CASKS
(17x17A01 @ 4.0% E)

Cask-to-Cask External Spacing (cm)					
External Moderator Density (%)	2	10	20	40	60
5	0.9352	0.9389	0.9356	0.9345	0.9351
10	0.9366	0.9353	0.9338	0.9357	0.9380
20	0.9368	0.9371	0.9359	0.9366	0.9372
50	0.9363	0.9363	0.9371	0.9352	0.9352
100	0.9355	0.9369	0.9354	0.9354	0.9354

Note:

1. All values are maximum k_{eff} which include the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0010.

Table 6.4.3

REACTIVITY EFFECTS OF SPACING AND WATER MODERATOR DENSITY FOR HEXAGONAL (TRIANGULAR-PITCHED) ARRAYS OF MPC-24 CASKS (17x17A01 @ 4.0% E)

	Cask-to-Cask External Spacing (cm)				
External Moderator Density (%)	2	10	20	40	60
5	0.9358	0.9365	0.9369	0.9354	0.9354
10	0.9363	0.9372	0.9351	0.9368	0.9372
20	0.9354	0.9357	0.9345	0.9358	0.9381
50	0.9347	0.9361	0.9371	0.9365	0.9370
100	0.9373	0.9381	0.9354	0.9354	0.9354

Note:

1. All values are maximum k_{eff} which include the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0009.

Table 6.4.4

REACTIVITY EFFECTS OF SPACING AND EXTERNAL MODERATOR DENSITY FOR HEXAGONAL (TRIANGULAR-PITCHED) ARRAYS OF MPC-24 CASKS (17x17A01 @ 4.0% E) INTERNALLY FLOODED WITH WATER OF 10% FULL DENSITY

	Cask-to-Cask External Spacing (cm)				
External Moderator Density (%)	2	10	20	40	60
10	0.4818	0.4808	0.4798	0.4795	0.4789
100	0.4798	0.4788	0.4781	0.4793	0.4793

Note:

1. All values are maximum k_{eff} which include the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0004 and 0.0005.

Table 6.4.5

CALCULATIONS FOR HEXAGONAL (TRIANGULAR-PITCHED) ARRAYS OF TOUCHING CASKS WITH MPC-24 AND MPC-68

MPC-24 (17x17A01 @ 4.0% ENRICHMENT)		
Internal Moderation (%)	External Moderation (%)	Maximum k_{eff}
0	0	0.3910
0	100	0.3767
100	0	0.9366
100	100	0.9341
MPC-68 (8x8C04 @ 4.2% ENRICHMENT)		
Internal Moderation (%)	External Moderation (%)	Maximum k_{eff}
0	0	0.4036
0	100	0.3716
100	0	0.9351
100	100	0.9340

Note:

1. All values are maximum k_{eff} which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0008 for 100% internal moderation, and between 0.0002 and 0.0003 for 0% internal moderation.

Table 6.4.6

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING FOR MPC-24 AND MPC-68

MPC-24 (17x17A01 @ 4.0% ENRICHMENT)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
MPC-68 (8x8C04 @ 4.2% ENRICHMENT)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348

Notes:

1. All values are maximum k_{eff} which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0010.

Table 6.4.7

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP FOR MPC-24 AND MPC-68

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded	0.9368	0.9348

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0010.

Table 6.4.8

MAXIMUM k_{eff} VALUES† IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Results					
	DFC Dimensions: ID 4.93" THK. 0.12"			DFC Dimensions: ID 4.81" THK. 0.11"		
	Max.†† k_{eff}	1 σ	EALF (eV)	Max.†† k_{eff}	1 σ	EALF (eV)
<u>6x6 Fuel Assembly</u>						
6x6 Intact Fuel	0.7086	0.0007	0.3474	0.7016	0.0006	0.3521
w/32 Rods Standing	0.7183	0.0008	0.2570	0.7117	0.0007	0.2593
w/28 Rods Standing	0.7315	0.0007	0.1887	0.7241	0.0006	0.1909
w/24 Rods Standing	0.7086	0.0007	0.1568	0.7010	0.0008	0.1601
w/18 Rods Standing	0.6524	0.0006	0.1277	0.6453	0.0007	0.1288
Collapsed to 8x8 array	0.7845	0.0007	1.1550	0.7857	0.0007	1.1162
Dispersed Powder	0.7628	0.0007	0.0926	0.7440	0.0007	0.0902
<u>7x7 Fuel Assembly</u>						
7x7 Intact Fuel	0.7463	0.0007	0.2492	0.7393	0.0006	0.2504
w/41 Rods Standing	0.7529	0.0007	0.1733	0.7481	0.0007	0.1735
w/36 Rods Standing	0.7487	0.0007	0.1389	0.7444	0.0006	0.1406
w/25 Rods Standing	0.6718	0.0007	0.1070	0.6644	0.0007	0.1082

† These calculations were performed with a planar-average enrichment of 3.0% and a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

†† Maximum k_{eff} includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.9

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Table 6.4.10

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Preferential Flooding	Fully Flooded
MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)	0.6560	0.7857
MPC-24E or MPC-24EF with 4 DFCs (Bounding All PWR Assembly Classes)	0.7895	0.9480

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.11

MAXIMUM k_{eff} VALUES IN THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt% ^{235}U.

Model Configuration inside the DFC	Maximum k_{eff}
Intact Assemblies (2 assemblies analyzed)	0.9340
Assemblies with missing rods (4 configurations analyzed)	0.9350
Collapsed Assemblies (6 configurations analyzed)	0.9360
Regular Arrays of Bare Fuel Rods (36 configurations analyzed)	0.9480

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.
2. The standard deviation (σ) of the calculations ranges between 0.0007 and 0.0010.

Table 6.4.12
SUMMARY OF THE CRITICALITY RESULTS FOR THE MOST REACTIVE ASSEMBLY FROM THE MOST REACTIVE ASSEMBLY CLASS IN EACH MPC-24
TO DEMONSTRATE COMPLIANCE WITH 10CFR71.55 AND 10CFR71.59

MPC-24, Assembly Class 15x15F, 4.1 wt% ^{235}U					
Configuration	% Internal Moderation	% External Moderation	Max.‡ k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	0.9410	0.0007	0.2998
Single Package, fully reflected	100%	100%	0.9397	0.0008	0.3016
Containment, fully reflected	100%	100%	0.9397	0.0008	0.3006
Infinite Array of Damaged Packages	100%	100%	0.9436	0.0009	0.2998
Infinite Array of Undamaged Packages	0%	0%	0.3950	0.0004	82612.0
MPC-68, Assembly Class 9x9E/F, 4.0 wt% ^{235}U					
Configuration	% Internal Moderation	% External Moderation	Max. k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	0.9486	0.0008	0.2095
Single Package, fully reflected	100%	100%	0.9470	0.0008	0.2079
Containment, fully reflected	100%	100%	0.9461	0.0007	0.2092
Infinite Array of Damaged Packages	100%	100%	0.9468	0.0008	0.2106
Infinite Array of Undamaged Packages	0%	0%	0.3808	0.0003	85218.0
MPC-68F, Assembly Class 6x6C, 2.7 wt% ^{235}U					
Configuration	% Internal Moderation	% External Moderation	Max. k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	0.8021	0.0007	0.2139
Single Package, fully reflected	100%	100%	0.8033	0.0008	0.2142
Containment, fully reflected	100%	100%	0.8033	0.0008	0.2138
Infinite Array of Damaged Packages	100%	100%	0.8026	0.0008	0.2142
Infinite Array of Undamaged	0%	0%	0.3034	0.0002	99463.0

‡ The maximum k_{eff} is equal to the sum of the calculated k_{eff}, two standard deviations, the code bias, and the uncertainty in the code bias.

Packages					

Table 6.4.12 (continued)
SUMMARY OF THE CRITICALITY RESULTS FOR THE MOST REACTIVE ASSEMBLY FROM THE MOST REACTIVE ASSEMBLY CLASS IN EACH MPC-24
TO DEMONSTRATE COMPLIANCE WITH 10CFR71.55 AND 10CFR71.59

MPC-24E/EF, Assembly Class 15x15F, 4.5 wt% ^{235}U					
Configuration	% Internal Moderation	% External Moderation	Max.‡ k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	0.9495	0.0008	0.3351
Single Package, fully reflected	100%	100%	0.9485	0.0008	0.3313
Containment, fully reflected	100%	100%	0.9486	0.0008	0.3362
Infinite Array of Damaged Packages	100%	100%	0.9495	0.0008	0.3335
Infinite Array of Undamaged Packages	0%	0%	0.4026	0.0004	87546.0
MPC-24E/EF TROJAN, Trojan Intact and Damaged Fuel, 3.7 wt% ^{235}U					
Configuration	% Internal Moderation	% External Moderation	Max. k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	0.9377	0.0008	n/c†
Single Package, fully reflected	100%	100%	0.9366	0.0008	n/c
Containment, fully reflected	100%	100%	0.9377	0.0008	n/c
Infinite Array of Damaged Packages	100%	100%	0.9383	0.0007	n/c
Infinite Array of Undamaged Packages	0%	0%	0.3518	0.0003	n/c
MPC-32, Assembly Class 15x15F / *17x17C*~~, 4.0 wt% ^{235}U~~					
Configuration	% Internal Moderation	% External Moderation	Max. k_{eff}	1 σ	EALF (eV)
Single Package, unreflected	100%	0%	*0.9480~~79~~*	*0.00054*	*0.4410~~37~~ ~~38~~*
Single Package, fully reflected	100%	100%	*0.9469~~78~~*	*0.0005*	*0.3480~~40~~ ~~17~~*
Containment, fully reflected	100%	100%	*0.94854*	*0.0004~~5~~*	*0.3476~~40~~ ~~17~~*
Infinite Array of Damaged Packages	100%	100%	*0.9475~~85~~*	*0.0004~~5~~*	*0.4402~~02~~ ~~0~~*

‡ The maximum k_{eff} is equal to the sum of the calculated k_{eff}, two standard deviations, the code bias, and the uncertainty in the code bias.
† n/c = not calculated

Infinite Array of Undamaged Packages	0%	0%	*0.4248~~11~~ ~~1~~*	*0.0002*	*45195~~443~~ ~~8~~*

Table 6.4.13

RESULTS FOR EACH ASSEMBLY CLASS IN THE MPC-24

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Max.† k_{eff}	1 σ	EALF (eV)
14x14A	4.6	0.9296	0.0008	0.2093
14x14B	4.6	0.9228	0.0008	0.2675
14x14C	4.6	0.9307	0.0008	0.3001
14x14D	4.0	0.8507	0.0008	0.3308
14x14E	5.0	0.7627	0.0007	0.3607
15x15A	4.1	0.9227	0.0007	0.2708
15x15B	4.1	0.9388	0.0009	0.2626
15x15C	4.1	0.9361	0.0009	0.2385
15x15D	4.1	0.9367	0.0008	0.2802
15x15E	4.1	0.9392	0.0008	0.2908
15x15F	4.1	0.9410	0.0007	0.2998
15x15G	4.0	0.8907	0.0008	0.3456
15x15H	3.8	0.9337	0.0009	0.2349
16x16A	4.6	0.9287	0.0008	0.2704
17x17A	4.0	0.9368	0.0008	0.2131
17x17B	4.0	0.9355	0.0008	0.2659
17x17C	4.0	0.9349	0.0009	0.2677

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

Table 6.4.14

RESULTS FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Max.† k_{eff}	1 σ	EALF (eV)
7x7B	4.2	0.9386	0.0007	0.3983
8x8B	4.2	0.9416	0.0007	0.3293
8x8C	4.2	0.9425	0.0007	0.3081
8x8D	4.2	0.9403	0.0006	0.2778
8x8E	4.2	0.9312	0.0008	0.2831
8x8F	4.0	0.9459	0.0007	0.2361
9x9A	4.2	0.9417	0.0008	0.2236
9x9B	4.2	0.9436	0.0008	0.2506
9x9C	4.2	0.9395	0.0008	0.2698
9x9D	4.2	0.9394	0.0009	0.2625
9x9E	4.0	0.9486	0.0008	0.2095
9x9F	4.0	0.9486	0.0008	0.2095
9x9G	4.2	0.9383	0.0008	0.2292
10x10A	4.2	0.9457	0.0008	0.2212
10x10B	4.2	0.9436	0.0007	0.2366
10x10C	4.2	0.9433	0.0007	0.2416
10x10D	4.0	0.9376	0.0008	0.3355
10x10E	4.0	0.9185	0.0007	0.2936

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

Table 6.4.15

RESULTS FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Max.† k_{eff}	1 σ	EALF (eV)
6x6A	2.7††	0.7888	0.0007	0.2310
6x6B†††	2.7	0.7824	0.0006	0.2184
6x6C	2.7	0.8021	0.0007	0.2139
7x7A	2.7	0.7974	0.0008	0.2015
8x8A	2.7	0.7697	0.0007	0.2158

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

†† These calculations were performed for 3.0% planar-average enrichment, however, the authorized contents are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the specification of authorized contents, Chapter 1.

Table 6.4.16

RESULTS FOR EACH ASSEMBLY CLASS IN THE MPC-24E/EF

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Max.† k_{eff}	1 σ	EALF (eV)
14x14A	5.0	0.9380	0.0008	0.2277
14x14B	5.0	0.9312	0.0008	0.2927
14x14C	5.0	0.9365	0.0008	0.3318
14x14D	5.0	0.8875	0.0009	0.4026
14x14E	5.0	0.7651	0.0007	0.3644
15x15A	4.5	0.9336	0.0008	0.2879
15x15B	4.5	0.9487	0.0009	0.3002
15x15C	4.5	0.9462	0.0008	0.2631
15x15D	4.5	0.9445	0.0008	0.3375
15x15E	4.5	0.9471	0.0008	0.3242
15x15F	4.5	0.9495	0.0008	0.3351
15x15G	4.5	0.9062	0.0008	0.3883
15x15H	4.2	0.9455	0.0009	0.2663
16x16A	5.0	0.9358	0.0008	0.3150
17x17A	4.4	0.9447	0.0007	0.2374
17x17B	4.4	0.9438	0.0008	0.2951
17x17C	4.4	0.9433	0.0008	0.2932

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

Table 6.4.17

RESULTS FOR THE MPC-24E/EF TROJAN

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Content	Max.† k_{eff}	1 σ	EALF (eV)
17x17B	3.7	Intact Fuel	0.9187	0.0009	not calculated
17x17B	3.7	Intact Fuel, Damaged Fuel and Fuel Debris	0.9377	0.0008	not calculated

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

Table 6.4.18

RESULTS FOR THE MPC-24E/EF TROJAN USING A FUEL FRAGMENT MODEL FOR DAMAGED FUEL AND FUEL DEBRIS

Fuel Cube OD (Inches)	Maximum k_{eff}			
	Fuel Volume / Water Volume			
	0.2	0.4	0.6	0.8
1	0.9098	0.9223	0.9260	0.9204
0.5	0.9156	0.9310	0.9273	0.9168
0.2	0.9254	0.9320	0.9216	0.9137
0.1	0.9253	0.9274	0.9183	0.9135
0.05	0.9224	0.9228	0.9168	0.9126
0.02	0.9183	0.9213	0.9140	0.9122

CHAPTER 8: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

8.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STAR 100 Package 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 Safety Analysis Report (SAR), the applicable regulatory requirements, and the Certificate of Compliance (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-STAR 100 Package will maintain containment of radioactive material; will maintain subcriticality control; will properly transfer the decay heat of the contained radioactive materials; and that radiation doses will meet regulatory requirements under all normal and hypothetical accident conditions of transport in accordance with 10CFR71 [8.0.1].

Both pre-operational and operational tests and inspections are performed throughout HI-STAR 100 loading operations to assure that the HI-STAR 100 Package is functioning within its design parameters. These include receipt inspections, nondestructive weld inspections, pressure tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 7 identifies the sequence of the tests and inspections. "Pre-operation", as referred to in this chapter, defines that period of time from receipt inspection of a HI-STAR 100 Package until the empty MPC is loaded into a HI-STAR overpack for fuel assembly loading.

The HI-STAR 100 Package is classified as important to safety (ITS). Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STAR 100 Package 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. Table 1.3.3 provides the safety classification and quality category, as applicable, for each major item or component of the HI-STAR 100 Package and required ancillary equipment and systems.

The acceptance criteria and maintenance program described in this chapter fully comply with the requirements of 10CFR Part 71.

8.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STAR 100 Package prior to or during use. These inspections and tests provide assurance that the HI-STAR 100 Package has been fabricated, assembled, inspected, tested, and accepted for use and loading under the conditions specified in this SAR and the Certificate of Compliance issued by the NRC in accordance with the requirements of 10CFR Part 71.

Noncompliances encountered during the required inspections and tests shall be corrected or dispositioned to bring the item into compliance with this SAR prior to use. Identification and resolution of noncompliances shall be performed in accordance with the Holtec International

Quality Assurance Program [8.1.1] or the licensee's NRC-approved Quality Assurance Program. The testing and inspection acceptance criteria applicable to the MPCs and the HI-STAR overpack are listed in Tables 8.1.1 and 8.1.2, respectively, and discussed in more detail in the sections that follow. These inspections and tests are intended to demonstrate that the HI-STAR 100 Package has been fabricated, assembled, and examined in accordance with the design evaluated in this SAR.

This section summarizes the test program established for the HI-STAR 100 Package.

8.1.1 Fabrication and Nondestructive Examination (NDE)

The design, material procurement, fabrication, and inspection of the HI-STAR 100 Package is performed in accordance with applicable codes and standards, including NRC-approved alternatives to the ASME Code, as specified in Tables 1.3.1 and 1.3.2, respectively, and on the drawings in Section 1.4. Additional details on specific codes used are provided below.

The following fabrication controls and required inspections shall be performed on the HI-STAR 100 Package, including the MPCs, in order to assure compliance with this SAR and the Certificate of Compliance.

1. Materials of construction specified for the HI-STAR 100 Package are identified in the drawings in Chapter 1. Important-to-safety materials shall be procured with certification and supporting documentation as required by ASME Code [8.1.2] Section II (when applicable); the applicable subsection of ASME Code Section III (when applicable); Holtec procurement specifications; and 10CFR71, Subpart H. 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 assure material traceability is maintained throughout fabrication for ITS items. Materials for the primary containment boundary of the HI-STAR overpack (bottom plate, inner shell, top flange, closure plate, port plugs, and closure plate bolts) and for the secondary containment boundary provided by the MPC (for the MPC-24EF and MPC-68F), shall also be inspected per the requirements of ASME Section III, Article NB-2500 Subsection NB.

2. The HI-STAR 100 Package primary containment boundary and the MPC (secondary containment boundary for MPC-24EF and MPC-68F) shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NB (see approved Code alternatives in Table 1.3.2). Other portions of the HI-STAR 100 Package shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NF (see approved Code alternatives in Table 1.3.2). The MPC basket and certain basket supports shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NG (see Tables 1.3.1, 1.3.2, and 1.3.3 for Code applicability and approved Code alternatives).

3. Welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable ASME Section III Subsections (e.g., NB, NG, or NF, as applicable to the SSC).

4. Welds shall be visually examined in accordance with ASME Code Section V, Article 9 with acceptance criteria per ASME Code Section III, Subsection NF, Article NF-5360, except the MPC fuel basket cell plate-to-cell plate welds and fuel basket support-to-canister welds, which shall have acceptance criteria to ASME Code Section III, Subsection NG, Article NG-5360, except as clarified by the Code alternatives in Table 1.3.2. Table 8.1.3 identifies additional nondestructive examination (NDE) requirements to be performed on specific welds, and the applicable codes and acceptance criteria to be used in order to meet the requirements of the applicable portions of Section III of the ASME Code. Acceptance criteria for NDE shall be in accordance with the applicable Code for which the item was fabricated, except as modified by the Code alternatives in Table 1.3.2. These additional NDE criteria are also specified in the drawings provided in Chapter 1 for the specific welds. Weld inspections shall be detailed in a weld inspection plan that identifies the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be reviewed and approved by Holtec International in accordance with its QA program. NDE inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [8.1.3] or other site-specific, NRC-approved program for personnel qualification.

5. The HI-STAR 100 system containment boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, 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 the effectiveness of the packaging.

6. Any welds requiring weld repair shall be repaired in accordance with the requirements of the ASME Code Section III, Article NB-4450, NG-4450, or NF-4450, as applicable to the SSC, and examined after repair in the same manner as the original weld.

7. Any base metal repairs shall be performed and examined in accordance with the applicable fabrication Code.

8. Grinding and machining operations of the HI-STAR 100 overpack primary containment boundary and the MPC shall be controlled through written and approved procedures and quality assurance oversight to ensure grinding and machining operations do not reduce base metal wall thicknesses of the boundaries beyond that allowed by the design. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets design requirements. A nonconformance shall be written for areas found to be below allowable base metal

thickness and shall be evaluated and repaired as necessary per the ASME Code Section III, Subsection NB requirements.

9. Dimensional inspections of the HI-STAR 100 Package shall be performed in accordance with written and approved procedures in order to verify compliance to design drawings and fit-up of individual components. All dimensional inspections and functional fit-up tests shall be documented.

10. All required inspections, examinations, and tests shall be documented. The inspection, examination, and test documentation shall become part of the final quality documentation package.

11. The HI-STAR 100 Package shall be inspected for cleanliness and proper preparation for shipping in accordance with written and approved procedures.

12. Each HI-STAR overpack shall be durably marked with the CoC identification number assigned by the NRC, trefoil radiation symbol, gross weight, model number, and unique identification serial number in accordance with 10CFR71.85(c) at the completion of the acceptance test program.

13. Deleted.

14. A completed quality documentation record package shall be prepared and maintained during fabrication of each HI-STAR 100 Package to include detailed records and evidence that the required inspections and tests have been performed for ITS items. The quality document record package shall be reviewed to verify that the HI-STAR 100 Package or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The quality documentation record package shall include, but not be limited to:

   - Completed Weld Records
   - Inspection Records
   - Nonconformance Reports
   - Material Test Reports
   - NDE Reports
   - Dimensional Inspection Reports

8.1.1.1 <u>MPC Lid-to-Shell Weld Volumetric Inspection</u>

1. The MPC lid-to-shell (LTS) weld (the confinement boundary closure per 10CFR72, and secondary containment (inner container) boundary per 10CFR71 for the MPC-68F and MPC-24EF) shall be volumetrically or multi-layer liquid penetrant examined following completion of field welding. If volumetric examination is used, the ultrasonic test (UT) method shall be employed. Ultrasonic techniques (including, as appropriate, Time-of-Flight Diffraction,

Focussed Phased Array, and conventional pulse-echo) shall be supplemented, as necessary, to ensure substantially complete coverage of the examination volume.

2. If volumetric examination is used, then a liquid penetrant (PT) examination of the root and final pass of the LTS weld shall be performed and unacceptable indications shall be documented, repaired and re-examined.

3. If a volumetric examination is not used, a multi-layer PT examination shall be employed. 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. The 3/8-inch weld depth corresponds to the maximum allowable flaw size.

4. The overall minimum thickness of the LTS weld has been increased by 0.125 inch over the size credited in the structural analyses to provide additional structural capacity (actual weld to be 0.75 inch for the standard MPC model and 1.25 inches for the "F" model). A J-groove weld 1/8" less was assumed in the structural analyses in Chapter 2.

5. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection process, including findings (indications) shall be made a permanent part of the 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 inspection of the weld shall be performed by qualified personnel and shall meet the acceptance requirements of ASME Section III, NB-5350 for PT and NB-5332 for UT.

6. Evaluation of any indications shall include consideration of any active flaw mechanisms. However, cyclic loading on the LTS weld is not significant, so fatigue will not be a factor. The LTS weld is protected from the external environment by the closure ring and the root of the LTS weld is dry and inert (He atmosphere), so stress corrosion cracking is not a concern for the LTS weld.

7. The volumetric or multi-layer PT examination of the LTS weld, in conjunction with other examinations that will be performed on this weld (PT of root and final pass, pressure test, and helium leakage test); the use of the ASME Code Section III acceptance criteria; and the additional 1/8th-inch of weld material conservatively not credited in the structural analyses, in total, provide reasonable assurance that the LTS weld is sound and will perform its secondary containment boundary function under all loading conditions. The volumetric (or multi-layer PT) examination and evaluation of indications provides reasonable assurance that

leakage of the weld or structural failure under normal or hypothetical accident conditions of transport will not occur.

8.1.2 Structural and Pressure Tests

8.1.2.1 Lifting Trunnions

Two trunnions (located near the top of the HI-STAR overpack) are provided for vertical lifting and handling of the HI-STAR 100 Package without the impact limiters installed. The trunnions are designed and shall be inspected and tested in accordance with ANSI N14.6 [8.1.5]. The trunnions are fabricated using a high-strength and high-ductility material (see overpack drawing in Section 1.4). The trunnions contain no welded components. The maximum design lifting load of 250,000 pounds for the HI-STAR 100 Package will occur during the removal of the HI-STAR overpack from the spent fuel pool after the MPC has been loaded, flooded with water, and the MPC lid is installed. The high material ductility, absence of materials vulnerable to brittle fracture, excellent stress margins, and a carefully engineered design to eliminate local stress risers in the highly-stressed regions (during lift operations) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612 [8.1.6], acceptance criteria for the lifting trunnions have been established in conjunction with other considerations applicable to heavy load handling.

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 (including trunnion 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 minimize the potential of load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load (750,000 lbs) shall be applied for a minimum of 10 minutes to the pair of lifting trunnions. The accessible parts of the trunnions (areas outside the HI-STAR overpack), and the local HI-STAR 100 cask areas shall then be visually examined to verify no deformation, distortion, or cracking has occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-STAR 100 cask areas shall require replacement of the trunnion and/or repair of the HI-STAR 100 cask. Following any replacements and/or repair, the load testing shall be re-performed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing shall be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section

II requirements provide further verification of the trunnion load capabilities. Test results shall be documented and shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above provide reasonable assurance that a handling accidents will not occur due to trunnion failure.

8.1.2.2 Pressure Testing

8.1.2.2.1 HI-STAR 100 Containment Boundary

The containment boundary of the HI-STAR Package shall be hydrostatically or pneumatically pressure tested to 150 psig +10,-0 psig, in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000. The test pressure of 150 psig is 150% of the Maximum Normal Operating Pressure (established per 10CFR71.85(b) requirements). This bounds the ASME Code Section III requirement (NB-6221) for hydrostatic testing to 125% of the design pressure (100 psig). The test shall be performed in accordance with written and approved procedures. The written and approved test procedure shall clearly define the test equipment arrangement.

The overpack pressure test may be performed at any time during fabrication after the containment boundary is complete. Preferably, the pressure test should be performed after overpack fabrication is complete, including attachment of the intermediate shells. The HI-STAR overpack shall be assembled for this test with the closure plate mechanical seal (only one required) or temporary test seal installed. Closure bolts shall be installed and torqued to a value less than or equal to the value specified in Table 7.1.3.

The calibrated test pressure gage installed on the overpack shall have an upper limit of approximately twice that of the test pressure. The test pressure shall be maintained for ten minutes. During this time period, the pressure gauge reading shall not fall below 150 psig. At the end of ten minutes, and while the pressure is being maintained at a minimum of 150 psig, the overpack shall be observed for leakage. In particular, the closure plate-to-top forging joint (the only credible leakage point) shall be examined. If a leak is discovered, the overpack shall be emptied and an evaluation shall be performed to determine the cause of the leakage. Repairs and retest shall be performed until the pressure test acceptance criterion is met.

After completion of the pressure testing, the overpack closure plate shall be removed and the internal surfaces shall be visually examined for cracking or deformation. Any evidence of cracking or deformation shall be cause for rejection or repair and retest, as applicable. The overpack shall be required to be pressure tested until the examinations are found to be acceptable.

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

8.1.2.2.2 MPC Secondary Containment Boundary

Pressure testing (hydrostatic or pneumatic) of the MPC secondary containment 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 the design pressure. The MPC vent and drain ports are used for pressurizing the MPC cavity. The loading procedures in Chapter 7 define the test equipment arrangement. The calibrated test pressure gage installed on the MPC pressure boundary shall have an upper limit of approximately twice that of the test pressure. Following completion of the required hold period at the test pressure, and after determining the leakage acceptance criterion is met, the surface of the MPC lid-to-shell weld shall be re-examined by liquid penetrant examination performed in accordance with ASME Code Section V, Article 6, with acceptance criteria per ASME Code Section III, Subsection NB, Article NB-5350. Any unacceptable areas shall require repair in accordance with the ASME Code Section III, Subsection NB, Article NB-4450. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. The performance and sequence of the test is described in Section 7.1 (loading procedures).

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 approved written procedures prepared in accordance with the ASME Code Section III, Subsection NB, NB-4450.

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

8.1.2.3 Materials Testing

The majority of materials used in the HI-STAR overpack are ferritic steels. ASME Code Section III and Regulatory Guides 7.11 [8.1.7] and 7.12 [8.1.8] require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures.

Each plate or forging for the HI-STAR 100 Package containment boundary (overpack inner shell, bottom plate, top flange, and closure plate) shall be required to be drop weight tested in accordance with the requirements of Regulatory Guides 7.11 and 7.12, as applicable. Additionally, per the ASME Code Section III, Subsection NB, Article NB-2300, Charpy V-notch testing shall be performed on these materials. Weld material used in welding the containment boundary shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NB, Articles NB-2300 and NB-2430.

Non-containment portions of the overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The non-containment materials to be tested include the intermediate shells, overpack port cover plates, and applicable weld materials.

Tables 2.1.22 and 2.1.23 provide the test temperatures or T_{NDT}, and test requirements to be used when performing the testing specified above.

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

8.1.2.4 Pneumatic Testing of the Neutron Shield Enclosure Vessel

A pneumatic pressure test of the neutron shield enclosure vessel shall be performed following final closure welding of the enclosure shell returns and enclosure panels. The pneumatic test pressure shall be 37.5+2.5,-0 psig, which is 125 percent of the relief device set pressure. The test shall be performed in accordance with approved written procedures.

During the test, the relief devices on the neutron shield enclosure vessel shall be removed. One of the relief device threaded connections is used for connection of the air pressure line and the other connection will be used for connection of the pressure gauge.

Following the introduction of pressurized gas into the neutron shield enclosure vessel, a 15 minute pressure hold time is required. If the neutron shield enclosure vessel fails to hold pressure, an approved soap bubble solution shall be applied to determine the location of the leak. The leak shall be repaired using weld repair procedures prepared in accordance with the ASME Code Section III, Subsection NF, Article NF-4450. The pneumatic pressure test shall be re-performed until no pressure loss is observed.

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

8.1.3 Leakage Testing

Leakage testing shall be performed in accordance with the requirements of ANSI N14.5 [8.1.9]. Testing shall be performed in accordance with written and approved procedures.

8.1.3.1 HI-STAR Overpack

A Containment System Fabrication Verification Leakage test of the welded structure shall be performed at any time after the containment boundary fabrication is complete. Preferably, this test should be performed at the completion of overpack fabrication, after all intermediate shells have been attached. The leakage test instrumentation shall have a minimum test sensitivity of 2.15×10^{-6} atm cm^3/s (helium). Containment boundary welds shall have indicated leakage rates not exceeding 4.3×10^{-6} atm cm^3/s (helium). If a leakage rate exceeding the acceptance criterion is detected, the area of leakage shall be determined using the sniffer probe method or other means, and the area shall be repaired per ASME Code Section III, Subsection NB, NB-4450 requirements. Following repair and appropriate NDE, the leakage testing shall be re-performed until the test acceptance criterion is satisfied.

At the completion of overpack fabrication, the total helium leakage through all helium retention penetrations (consisting of the inner mechanical seal between the closure plate and the top flange and the vent and drain port plug seals) shall be demonstrated to not exceed the leakage rate of 4.3×10^{-6} atm cm^3/sec (helium) at a minimum test sensitivity of 2.15×10^{-6} atm cm^3/sec (helium). This may be performed simultaneously with the Containment System Fabrication Verification Leakage test or may be performed separately using the methods described in the paragraph below.

At the completion of fabrication, a Containment System Fabrication Verification Leakage test shall be performed on the HI-STAR overpack closures. Helium leakage through the containment penetrations (consisting of the inner mechanical seal between the closure plate and top flange, and the vent and drain port plug seals) shall be demonstrated to not exceed a leakage rate of 4.3×10^{-6} atm cm^3/s (helium) at a minimum test sensitivity of 2.15×10^{-6} atm cm^3/s (helium).

The leakage testing of the penetrations is performed by evacuating and backfilling the overpack with helium gas to an appropriate pressure. A helium Mass Spectrometer Leak Detector (MSLD) with a minimum calibrated sensitivity of 2.15×10^{-6} atm cm^3/s (helium) shall be used in parallel with a vacuum pump and a test cover (see Chapter 7 for details) designed for testing the penetration seals. The test cover is connected. The cavity on the external side of the port plug to be tested is evacuated and the vacuum pump is valved out. The MSLD detector measures the leakage rate of helium into the test cavity. If the leakage rate exceeds a leakage rate of 4.3×10^{-6} atm cm^3/s (helium), the test chamber is vented and removed. The corresponding plug seal is removed, seal seating surfaces are inspected and cleaned, and the plug with a new seal is reinstalled and torqued to the required value. The test process is then repeated until the seal leakage rate is successfully achieved. The same process is repeated for the remaining overpack vent or drain port. The process is used for the closure plate seals except the closure plate test tool (see Chapter 7 for details) is used in lieu of the test cover.

If the total measured leakage rate for all tested penetrations does not exceed 4.3×10^{-6} atm cm^3/sec, the leakage tests are successful. If the total leakage rate exceeds 4.3×10^{-6} atm cm^3/sec, an evaluation should be performed to determine the cause of the leakage, repairs made as necessary, and the overpack must be re-tested until the total leakage rate is within the required acceptance criterion. Leak testing results for the HI-STAR overpack shall become part of the quality record documentation record package.

8.1.3.2 MPC Secondary Containment Boundary

After the completion of welding the MPC shell to the baseplate, a confinement boundary weld leakage test shall be performed using a helium MSLD as described in Chapter 7.. These leakage tests are performed on all MPCs as a good practice to confirm the CoC leakage rate limits are not exceeded. However, the MPC only performs a secondary containment function for MPC-68F and MPC-24EF, which transport fuel debris. The MPC leakage test used to demonstrate compliance with the CoC leakage acceptance criterion for MPC-68F and MPC-24EF is performed prior to shipment as later in this section. The pressure boundary welds of the MPC canisters shall have indicated leakage rates not exceeding $5x10^{-6}$ atm cm^3/s (helium) with a minimum test sensitivity of $2.5x10^{-6}$ atm cm^3/sec (helium). If leakage rates exceeding the test criteria are detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, NB-4450, requirements. Re-testing of the MPC shall be performed until the leakage rate acceptance criterion is met.

Leakage testing of the field welded MPC lid-to-shell weld shall be performed following completion of the MPC pressure test performed per Subsection 8.1.2.2.2. 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 tests are provided in Section 7.1.

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

Prior to the transport of an MPC-68F or MPC-24EF containing fuel debris in the HI-STAR 100 Package, a Containment Fabrication Verification Leakage Test shall be performed on the secondary containment boundary of the MPC. The test is performed with the MPC loaded into the HI-STAR overpack. The HI-STAR overpack annulus is sampled to inspect for radioactive material and then evacuated to an appropriate vacuum condition. The HI-STAR overpack annulus is then isolated from the vacuum pump. Following an appropriate isolation period, the HI-STAR overpack annulus atmosphere is sampled for helium leakage from the MPC. The test is considered acceptable if the detected leakage from the MPC does not exceed $5x10^{-6}$ atm cm^3/s (helium) with a test sensitivity of $2.5x10^{-6}$ atm cm^3/s (helium). If the acceptance criterion is not met, transport of the MPC-68F or MPC-24EF is not authorized. Corrective actions from re-testing, up to and including off-loading of the MPC, shall be taken until the leakage rate acceptance criterion is met.

8.1.4 Component Tests

8.1.4.1 Valves, Relief Devices, and Fluid Transport Devices

There are no fluid transport devices associated with the HI-STAR 100 Package. The only valve-like components in the HI-STAR 100 Package 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 leak tested to verify the MPC secondary containment (MPC-68F and MPC-24EF) boundary.

The vent and drain ports in the HI-STAR overpack are accessed through port plugs specially designed for removal and installation using connector tools. The tools are described and presented in figures in Chapter 7.

There are two relief devices (e.g., rupture discs) installed in the upper ledge surface of the neutron shield enclosure vessel of the HI-STAR overpack. These relief devices are provided for venting purposes under hypothetical fire accident conditions in which vapor formation from neutron shielding material degradation may occur. The relief devices are designed to relieve at 30 psig (± 5 psig).

8.1.4.2 Seals and Gaskets

Two concentric mechanical seals are provided on the HI-STAR overpack closure plate to provide containment boundary sealing. Mechanical seals are also used on the overpack vent and drain port plugs of the HI-STAR overpack containment boundary. Each primary seal is individually leak tested in accordance with Subsection 8.1.3.1. prior to the HI-STAR 100 Package's first use and during each loading operation. An independent and redundant seal is provided for each penetration (e.g., closure plate, port cover plates, and closure plate test plug). No containment credit is taken for these redundant seals and they are not leakage tested. Details on these seals are provided in Chapter 4.

8.1.4.3 Transport Impact Limiter

The removable HI-STAR transport impact limiters consist of aluminum honeycomb crush material arranged around a carbon steel structure and enclosed by a stainless steel shell. The drawings in Chapter 1 specify the crush strength of the aluminum honeycomb materials (nominal +/- 7%) for each zone of the impact limiter. For manufacturing purposes, verification of the impact limiter material is accomplished by performance of a crush test of sample blocks of aluminum honeycomb material for each large block manufactured. The verification tests are performed by the aluminum honeycomb supplier in accordance with approved procedures. The certified test results shall be submitted to Holtec International with each shipment.

All welds on the HI-STAR impact limiter shall be visually examined in accordance with the ASME Code, Section V, Article 9, with acceptance criteria per ASME Section III, Subsection NF, Article NF-5360.

8.1.5 Shielding Integrity

The HI-STAR 100 System has three specifically designed shields for neutron and gamma ray attenuation. For gamma shielding, there are successive carbon steel intermediate shells attached onto the outer surface of the overpack inner shell. The details of the manufacturing process are discussed in Chapter 1. Holtite-A neutron shielding is provided in the outer enclosure of the overpack. Additional neutron attenuation is provided by the encased Boral neutron absorber attached to the fuel basket cell surfaces inside the MPCs. Test requirements for each of the three shielding items are described below.

8.1.5.1 Fabrication Testing and Controls

Holtite-A:

Neutron shield properties of Holtite-A are provided in Chapter 1. Each manufactured lot of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet the requirements specified in Chapter 1. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration, and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality record documentation package.

The installation of the neutron shielding material shall be performed in accordance with written, approved, and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps or voids from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International.

Steel:

The steel plates utilized in the construction of the HI-STAR 100 Package shall be dimensionally inspected to assure compliance with the drawings in Section 1.4.

The total measured thickness of the inner shell plus intermediate shells shall be nominally 8.5 inches over the total surface area of the overpack shell. The top flange, closure plate, and bottom plate of the overpack shall be measured to confirm their thicknesses meet drawing requirements of Section 1.4. Measurements shall be performed in accordance with written and approved procedures. Measurements shall be made through a combination of receipt inspection thickness measurements on individual plates and actual measurements taken prior to welding the forgings and shells. Any area found to be under the specified minimum thickness shall be repaired in accordance with applicable ASME Code requirements.

No additional gamma shield testing of the HI-STAR 100 Package is required. A shielding effectiveness test as described in Subsection 8.1.5.2 shall be performed on each fabricated HI-STAR 100 Package after the first fuel loading.

General Requirements for Shield Materials:

1. Test results shall be documented and become part of the quality documentation package.

2. Dimensional inspections of the cavities containing poured neutron shielding materials shall assure that the amount of shielding material specified in the design documents is incorporated into the fabricated item.

8.1.5.2 Shielding Effectiveness Tests

Users shall implement procedures which verify the integrity of the Holtite-A neutron shield once for each overpack. Neutron shield integrity shall be verified via measurements either at first use or with a check source using, at a maximum, a 6x6 inch test grid over the entire surface of the neutron shield, including the impact limiters.

Following the first fuel loading of each HI-STAR 100 Package, a shielding effectiveness test shall be performed to verify the effectiveness of the neutron shield. This test shall be performed either with a check source or with loaded contents. If the test is performed using loaded contents, the test shall be performed after the HI-STAR 100 Package has been, drained, sealed, and backfilled with helium.

The shielding effectiveness tests shall be performed using written and approved procedures. Calibrated radiation detection equipment shall be used to take measurements at the surface of the HI-STAR overpack. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. Measurements shall be documented and become part of the quality documentation package. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.

8.1.5.3 Neutron Absorber Tests

Each plate of Boral shall be visually inspected by the manufacturer for damage (e.g., scratches, cracks, burrs, and peeled cladding) and foreign material embedded in the surfaces. In addition, the MPC fabricator shall visually inspect the Boral plates on a lot sampling basis. The sample size shall be determined in accordance with MIL-STD-105D or equivalent. The selected Boral plates shall be inspected for damage such as inclusions, cracks, voids, delamination, and surface finish.

NOTE

The following two paragraphs are incorporated by reference into the HI-STAR 100 System Part 71 CoC (Section 6.b.9) by reference and may not be deleted or altered in any way without prior NRC approval via CoC amendment. The texts of these paragraphs are, therefore, shown in bold type to distinguish them from other text.

After manufacturing, a statistical sample of each lot of Boral shall be tested using wet chemistry and/or neutron attenuation techniques to verify a minimum ^{10}B content at the ends of the panel. The minimum ^{10}B loading of the Boral panels for each MPC model is

provided in Table 1.2.3 Any panel in which ^{10}B loading is less than the minimum allowed per the drawings in Section 1.4 shall be rejected.

Tests shall be performed using written and approved procedures. Results shall be documented and become part of the quality records documentation package.

Installation of Boral panels into the fuel basket shall be performed in accordance with written and approved procedures (or shop travelers). Travelers and/or quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a Boral panel in accordance with the drawings in Section 1.4. These quality control processes, in conjunction with Boral manufacturing testing, provide the necessary assurances that the Boral will perform its intended function. The criticality design for the HI-STAR 100 System is based on favorable geometry and fixed neutron poisons. The inert helium environment inside the MPC cavity where the Boral is located ensures that the poisons will remain effective for the life of the canister. Given the design and service conditions, there are no credible means to lose the fixed neutron poisons. Therefore, no additional testing is required to ensure the Boral is present and in proper condition per 10 CFR 71.87(g).

8.1.6 Thermal Acceptance Test

The first fabricated HI-STAR overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the drawings in Section 1.4. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures.

Steam heating of the overpack cavity surfaces is the preferred method for this test instead of electric heating. There are several advantages with steam heated testing as listed below:

(i) Uniform cavity surface temperatures are readily achieved as a result of high steam condensation heat transfer coefficient (about 2,000 Btu/ft^2 hr-°F compared to about 1 Btu/ft^2 hr-°F for air) coupled with the steam's uniform distribution throughout the cavity.

(ii) A reliable constant temperature source (steam at atmospheric pressure condenses at 212°F compared to variable heater surface temperatures in excess of 1,000°F) eliminates concerns of overpack cavity surface overheating.

(iii) Interpretation of isothermal test data is not susceptible to errors associated with electric heating systems due to heat input measurement uncertainties, leakage of heat from electrical cables, thermocouple wires, overpack lid, bottom baseplate, etc.

(iv) The test setup is simple requiring only a steam inlet source and drain compared to numerous power measurement and control instruments, switchgear and safety interlocks required to operate an electric heater assembly.

Twelve (12) calibrated thermocouples shall be installed on the external walls of the overpack as shown in Figure 8.1.2. Three calibrated thermocouples shall be installed on the internal walls of the overpack in locations to be determined by procedure. Additional temperature sensors shall be used to monitor ambient temperature, steam supply temperature, and condensate drain temperature. The thermocouples shall be attached to strip chart recorders or other similar mechanism to allow for continuous monitoring and recording of temperatures during the test. Instrumentation shall be installed to monitor overpack cavity internal pressure.

After the thermocouples have been installed, dry steam will be introduced through an opening in the test cover plate previously installed on the overpack and the test initiated. Temperatures of the thermocouples, plus ambient, steam supply, and condensate drain temperature shall be recorded at hourly intervals until thermal equilibrium is reached. Appropriate criteria defining when thermal equilibrium is achieved shall be determined based on a variety of potential ambient test conditions and incorporated into the test procedure. In general, thermal equilibrium is expected approximately 12 hours after the start of steam heating. Air will be purged from the overpack cavity via venting during the heatup cycle. During the test, the steam condensate flowing out of the overpack drain shall be collected and the mass of the condensate measured with a precision weighing instrument.

Once thermal equilibrium is established, the final ambient, steam supply, and condensate drain temperatures and temperatures at each of the thermocouples shall be recorded. The strip charts, hand-written logs, or other similar readout shall be marked to show the point when thermal equilibrium was established and final test measurements were recorded. The final test readings along with the hourly data inputs and strip charts (or other similar mechanism) shall become part of the quality records documentation package for the HI-STAR 100 Package.

The heat rejection capability of the overpack at test conditions shall be computed using the following formula:

$$Q_{hm} = (h_1 - h_2)\, m_c \tag{8-1}$$

Where:

Q_{hm} = Heat rejection rate of the overpack (Btu/hr)

h_1 = Enthalpy of steam entering the overpack cavity (Btu/lbm)

h_2 = Enthalpy of condensate leaving the overpack cavity (Btu/lbm)

m_c = Average rate of condensate flow measured during thermal equilibrium conditions (lbm/hr)

Based on the HI-STAR 100 overpack thermal model, a design basis minimum heat rejection capacity (Q_{hd}) shall be computed at the measured test conditions (i.e., steam temperature in the overpack cavity and ambient air temperature). The thermal test shall be considered acceptable if the measured heat rejection capability is greater than the design basis minimum heat rejection capacity ($Q_{hm} > Q_{hd}$).

The summary of reference ambient inputs that define the thermal test environment are provided in Table 8.1.4. In Figure 8.1.3, a steady-state temperature contour plot of a steam heated overpack is provided based on the thermal analysis methodology described in SAR Chapter 3. Transient heating of the overpack is also determined to establish the time required to approach (within 2° F) the equilibrium temperatures. The surface temperature plot shown in Figure 8.1.4 demonstrates that a 12-hour steam heating time is adequate to closely approach the equilibrium condition.

If the acceptance criteria above are not met, then the HI-STAR 100 Package shall not be accepted until the root cause is determined, appropriate corrective actions are completed, and the overpack is re-tested with acceptable results.

Test results shall be documented and shall become part of the quality record documentation package.

8.1.7 <u>Cask Identification</u>

Each HI-STAR 100 Package shall be provided with unique identification plates with appropriate markings per 10CFR71.85(c) and 10CFR72.236(k). The identification plates shall not be installed until each HI-STAR 100 Package component has completed the fabrication acceptance test program and been accepted by authorized Holtec International personnel. A unique identifying serial number shall also be stamped on the MPC to provide traceability back to the MPC specific quality records documentation package.

Table 8.1.1

MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of MPC components per ASME Code Section III, Subsections NB, NF, and NG, , per NB-5300, NF-5300, and NG-5300, as applicable. b) A dimensional inspection of the fuel 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 drawings using standard American Welding Society NDE symbols and/or notations. 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 8.1.1 (continued)

MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of MPC components shall be performed per ASME Code Section IX and III, Subsections NB, NF, and NG, as applicable. b) Materials analysis (steel, Boral, etc.), shall be performed and records shall be kept in a manner commensurate with "important to safety" classifications.	a) None.	a) An ultrasonic (UT) examination or multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld shall be performed per ASME Section V, Article 5 (or ASME Section V, Article 2). Acceptance criteria for the examination are defined in Subsection 8.1.1.1. b) ASME Code NB-6000 pressure test shall be performed after MPC closure welding. Acceptance criteria are defined in Subsection 8.1.2.2.2.
Leak Tests	a) Helium leak rate testing shall be performed on all MPC pressure boundary shop welds.	a) None.	a) Helium leak rate testing shall be performed on MPC lid-to-shell, and vent and drain ports-to-MPC lid field welds after closure welding. Acceptance criteria are defined in Subsection 8.1.3.2. b) A Containment System Fabrication Verification Leakage Test shall be performed on the MPC-68F and MPC-24EF prior to the transport of the HI-STAR 100 Package containing fuel debris. Acceptance criteria are defined in Subsection 8.1.3.2.

Table 8.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Criticality Safety	a) The boron content shall be verified at the time of neutron absorber material manufacture. b) The installation of Boral panels into MPC basket plates shall be verified by inspection.	a) None.	a) None.
Shielding Integrity	a) Material compliance shall be verified through CMTRs. b) Dimensional verification of MPC lid thickness shall be performed.	a) None.	a) None.
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is to be tested 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 to be 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 8.1.2

HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of the HI-STAR overpack shall be performed per ASME Code, Subsection NB, NB-5300 for containment boundary components, and Subsection NF, NF-5300 for non-containment boundary components. b) A dimensional inspection of the overpack internal cavity, external dimensions, and closure plate shall be performed to verify compliance with design requirements. c) The HI-STAR overpack system containment boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, 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 the effectiveness of the packaging. d) NDE of weldments shall be defined on drawings using standard American Welding Society NDE symbols and/or notations. e) Cleanliness of the HI-STAR overpack shall be verified upon completion of fabrication. f) Packaging of the HI-STAR overpack at the completion of fabrication shall be verified prior to shipment. g) Examination of the AL-STAR impact limiters shall be performed per ASME Code, Subsection NF, NF-5300.	a) The HI-STAR overpack shall be visually inspected prior to placement in service at the licensee's facility. b) HI-STAR overpack protection at the licensee's facility shall be verified. c) HI-STAR overpack cleanliness and exclusion of foreign material shall be verified prior to use.	a) Inspect overpack cavity and accessible external surfaces prior to each fuel loading. b) Visually inspect impact limiters for damage and compliance to drawing requirements prior to each transport.

Table 8.1.2 (continued)

HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of HI-STAR overpack components shall be performed per ASME Code, Subsection NB and NF, as applicable. b) 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 classification category. c) A load test of the lifting trunnions shall be performed during fabrication. d) A pressure test of the containment boundary in accordance with ASME Code Section III, Subsection NB-6000 and 10CFR71.85(b) shall be performed. e) A pneumatic pressure test of the neutron shield enclosure shall be performed during fabrication.	a) None.	a) The relief devices on the neutron shield vessel shall be replaced every 5 years.

Table 8.1.2 (continued)

HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Leak Tests	a) Containment Fabrication Verification Leakage rate testing of the HI-STAR containment boundary welds shall be performed in accordance with ANSI N14.5. b) A Containment System Fabrication Verification Leakage rate test shall be performed on all HI-STAR overpack containment boundary mechanical seal boundaries in accordance with ANSI N14.5 at the completion of fabrication.	a) None.	a) Containment System Periodic Verification Leakage Test of the HI-STAR 100 Package shall be performed prior to each loaded transport (if not previously tested within 12 months). b) Containment System Fabrication Verification Leakage Test of the HI-STAR 100 Package shall be performed after the third use.
Criticality Safety	a) None.	a) None.	a) None.
Shielding Integrity	a) Material verifications (Holtite-A, shell plates, etc.), shall be performed in accordance with the item's safety classification. The required material certifications shall be obtained. b) The placement of Holtite-A shall be controlled through written special process procedures.	a) None.	a) A shielding effectiveness test of the neutron shield shall be performed every five years while in service. b) Verify the integrity of the Holtite-A neutron shield once at first use or with a check source.

Table 8.1.2 (continued)

HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operation
Thermal Acceptance	a) A thermal acceptance test is performed at completion of fabrication of the first HI-STAR overpack to confirm the heat transfer capabilities.	a) None.	a) An in-service thermal test shall be performed every five years during transport operations, or prior to transport if period exceeds five years from previous test. Acceptance criteria are defined in Section 8.2.6.
Cask Identification Inspection	a) Identification plates shall be installed on the HI-STAR overpack at completion of the acceptance test program.	a) The identification plates shall be checked prior to loading.	a) The identification plates shall be periodically inspected per licensee procedures and shall be repaired or replaced if damaged.
Fit-Up Tests	a) Fit-up tests of HI-STAR 100 Package components (closure plates, port plugs, cover plates impact limiters (if available)), shall be performed during fabrication.	a) Fit-up test of the HI-STAR overpack lifting trunnions with the lifting yoke shall be performed. b) Deleted c) Fit-up test of the MPC into the HI-STAR overpack shall be performed prior to loading.	a) Fit-up of all removable components shall be verified during each loading operation.

Table 8.1.3 HI-STAR 100 NDE REQUIREMENTS			
MPC			
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 or UT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 5 (UT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 UT: ASME Section III, Subsection NB, Article NB-5330 PT: ASME Section III, Subsection NB, Article NB-5350

Table 8.1.3 (continued)

HI-STAR 100 NDE REQUIREMENTS

MPC			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Lid-to-shell	PT (root and final pass) and multi-layer PT (if UT is not performed). PT (surface following pressure test) UT (if multi-layer PT is not performed)	ASME Section V, Article 6 (PT) ASME Section V, Article 5 (UT)	PT: ASME Section III, Subsection NB, Article NB-5350 UT: ASME Section III, Subsection NB, Article NB-5332
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
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 8.1.3 (continued) HI-STAR 100 NDE REQUIREMENTS			
HI-STAR OVERPACK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Inner shell-to-top flange	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Inner shell-to-bottom plate	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Inner shell longitudinal seam	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350

Table 8.1.3 (continued) HI-STAR 100 NDE REQUIREMENTS			
HI-STAR OVERPACK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Inner shell circumferential seam	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Intermediate shell welds (as noted on drawings)	MT or PT (surface)	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT ASME Section III, Subsection NF, Article NF-5340

Table 8.1.4

SUMMARY OF OVERPACK THERMAL ANALYSIS
REFERENCE AMBIENT INPUTS

PARAMETER	VALUE
Steam Temperature	212°F
Ambient Temperature	70°F
Radiative Blocking	None[1]
Exposed Surfaces Insolation	None

1 The test shall be performed on an isolated overpack. Thus, cask radiation blocking at an ISFSI array is not applicable to test conditions.

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(50 pages plus this cover sheet)

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 24

MODEL NO. HI-STAR 100 SYSTEM

NRC FORM 618
(3-96)
10cfr71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1.a CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9261	24	USA/9261/B(U)F-85	1	11

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of and country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO (*Name and Address*)	b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:
Holtec International Holtec Center 555 Lincoln Drive West Marlton, NJ 08053	Holtec International application dated October 23, 1995, as supplemented c. DOCKET NUMBER 71-9261

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10CFR Part 71, as applicable, and the conditions specified below.

5.

5.a. Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses for the package are called out in drawings listed below.

Multi-Purpose Canister

There are ~~five~~ *six* Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-24E, MPC-24EF, *MPC-32,* MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions, except those MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design. A single overpack design is provided that is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designe to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; *the MPC-32 is*

5. a. (2) Description (continued)

designed to contain up to 32 PWR fuel assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 loaded with material classified as fuel debris is designated as MPC-68F. Any MPC-24E loaded with material classified as fuel debris is designated as MPC-24EF.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. The fuel basket designs vary based on the MPC model. For the HI-STAR 100 System transporting fuel debris in a MPC-68F or MPC-24EF, the MPC provides the second inner container, in accordance with 10CFR71.63. The MPC pressure boundary is a strength-welded enclosure constructed entirely of a stainless steel alloy.

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12~~10~~:

(a) HI-STAR 100 MPC-24 Fuel Basket	Drawing 3926, Sheets 1-4, Rev. 5
(b) HI-STAR 100 MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 4

5. a. (3) Drawings (continued)

(c) HI-STAR 100 MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 4
(d) HI-STAR 100 MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. 8
(e) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 5
(f) HI-STAR 100 Impact Limiters	Drawing C1765, Sheets 1-6, Rev. 1; and Sheet 7, Rev. 0
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 1
(h) Trojan MPC Spacer	Drawing 4111, Sheets 1 and 2, Rev. 0
(i) Trojan Failed Fuel Can	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 7
(j) Trojan Failed Fuel Can Spacer	Drawing 4122, Sheets 1 and 2, Rev. 0
(k) Holtec Damaged Fuel Container for Trojan plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(l) HI-STAR 100 MPC-32 Fuel Basket	*Drawing 3927, Sheets 1-4, Rev. 6*

5. b. Contents

(1) Type and Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(g) below are authorized for transportation.

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

5.b.1.(b) Definitions (continued)

Damaged Fuel Containers (or Canisters) (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10 and 1.2.11 of the HI-STAR 100 System SAR, Rev. ~~10~~*12*.

Fuel Debris is ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 0.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

5.b.1.(b) Definitions (continued)

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR 100 System SAR, Rev. ~~10~~*12*.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. ~~10~~*12*.

ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the two limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPC-68s and MPC-68Fs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of th two limits for the damaged fuel assemblies or the intact fuel assemblies.

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining Zircaloy-clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies or the applicable Zircaloy-clad fuel assemblies.

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For the MPC-32, the administrative procedures to ensure that the cask will be loaded with fuel that is within the specifications should include a measurement that confirms the reactor record for each assembly. Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that measure each assembly.

The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For confirmation of assembly reactor burnup record(s), the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement.

c. Transport Index for Criticality Control

The minimum transport index to be shown on the label for nuclear criticality control: 0

6. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the following provisions:

6.a (continued)

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b above.

(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.

(3) The package must satisfy the following leak testing requirements:

(a) All overpack containment boundary seals shall be leak tested to show a total leak rate of not greater than 4.3×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} atm cm^3/sec (helium) and shall be performed:

(i) within the 12-month period prior to each shipment;
(ii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iii) After each seal replacement.

(b) Within 30 days before each shipment, all overpack containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a(3)(a) above.

(c) Each overpack containment boundary seal must be replaced after each use of the seal.

(4) The relief devices on the neutron shield vessel shall be replaced every 5 years.

(5) MPC-68F and MPC-24EF shall be leak tested prior to shipment to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} atm cm^3/sec (helium).

(6) MPCs deployed at an ISFSI under 10 CFR 72 prior to transportation may be dried using the vacuum drying method or the Forced Helium Dehydration (FHD) method. MPCs placed directly into transportation service under 10 CFR 71 without first being deployed at an ISFSI must be dried using the FHD method. Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

6.a.(6) (continued)

For those MPCs vacuum dried:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

For those MPCs dried using the FHD System:

(a) Following bulk moisture removal, the temperature of the gas exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes.

(7) Following drying, the MPC shall be backfilled with 99.995% minimum purity helium: > 0 psig and ≤ 44.8 psig at a reference temperature of 70°F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The overpack annulus shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack annulus shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(9) Following vacuum drying, the overpack shall be backfilled with helium to ≥ 10 psig and ≤ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2985 ± 90
Overpack Vent and Drain Port Plugs	45 +5/-2
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.

b. All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed and shall include the following provisions:

(1) The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.

6.b (continued)

(2) The MPC shall be pressure tested in accordance with ASME Section III, Subsection NB, Article NB-6110 and applicable sub-articles. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of the design pressure. The minimum hydrostastic test pressure shall be 125 psig. If pneumatic testing is used, the MPC shall be pressure tested to 120% of the design pressure. The minimum pneumatic test pressure shall be 120 psig.

(3) The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the Ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection results, including all relevant indications shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(5) The radial neutron shield shall have a minimum thickness of 4.3 inches and the impact limiter neutron shields shall have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity shall be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Measurements shall be performed over the entire exterior surface of the radial neutron shield and each impact limiter using, at a maximum, a 6 x 6 inch test grid.

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years of each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 1.4 of the SAR, Rev. 9. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

6.b (continued)

(8) For each package, a periodic thermal performance test shall be performed every 5 years or prior to next use, if the package has not been used for transport for greater than 5 years, to demonstrate that the thermal capabilities of the cask remain within its design basis.

(9) The MPC neutron absorber's minimum acceptable ^{10}B loading is 0.0267 g/cm^2 for the MPC-24 and 0.0372 g/cm^2 for the MPC-24E, MPC-24EF, *MPC-32*, and MPC-68; and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques. *The neutron absorber test requirements in Section 8.1.5.3 of the HI-STAR 100 SAR are hereby incorporated by reference into this CoC.*

(10) a. The minimum flux trap size for the MPC-24 is 1.09 inches.

b. The minimum flux trap sizes for the generic MPC-24E and MPC-24EF are 0.776 inch for cells 3, 6, 9,and 22; and 1.076 inch for the remaining cells.

c. The minimum flux trap sizes for the Trojan MPC-24E and MPC-24EF are 0.526 inch for cells 3, 6, 9,and 22; and 1.076 inch for the remaining cells.

(11) *a.* The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

b. The minimum fuel cell pitch for the MPC-32 is 9.158 inches.

(12) The package containment verification leak test shall be per ANSI N14.5-1997.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

9. The personnel barrier shall be installed at all times while transporting a loaded overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration Date: ~~March 31, 2004~~*TBD*

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision ~~10~~*12*, dated *TBD*.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: TBD

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Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels.
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.

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Table A.1 *(Page 1 of 20)*
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly	
i. ZR Clad:	≤ 833 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (*Page 2 of 20*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U and (2) array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, and a minimum initial enrichment $\geq$ 2.4 wt% ^{235}U.
ii. SS Clad:	An assembly cooling time after discharge $\geq$ 16 years, an average burnup $\leq$ 22,500 MWD/MTU, and a minimum initial enrichment $\geq$ 3.5 wt% ^{235}U.
e. Decay heat per assembly	
i. ZR Clad:	$\leq$ 272 Watts, except for array/class 8x8F fuel assemblies, which shall have a decay heat $\leq$ 183.5 Watts
ii. SS Clad:	$\leq$ 83 Watts
f. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
g. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
h. Fuel assembly weight	$\leq$ 700 lbs, including channels

Table A.1 (*Page 3 of 20*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (*Page 4 of 20*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum Planar-Average Initial Enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table A.1 (*Page 5 of 20*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight	≤ 550 lbs, including channels and damaged fuel container

Table A.1(Page 6 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods Per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel Pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 7 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight	≤ 400 lbs, including channels

Table A.1 (*Page 8 of 20*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (*Page 9 of 20*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (*Page 10 of 20*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table A.1 (*Page 11 of 20*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (*Page 12 of 20*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTIHM, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight	≤ 550 lbs, including channels and damaged fuel container

Table A.1(Page 13 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1(Page 14 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1(Page 15 of 20)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR Clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv. Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup ss specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1(Page 16 of 20)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan plant fuel.

F. One (1) Trojan plant Sb-Be and/or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1(Page 17 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. Zr Clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv. Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup ss specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1(Page 18 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

Table A.1(Page 19 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs, including non-fuel hardware and DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be transported in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan plant fuel.

F. One (1) Sb-Be and/or two (2) Cf neutron sources may be transported. Each neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly or fuel classified as fuel debris.

Table A.1(Page 20 of 20)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, maximum and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly	Calculated value as a function of initial enrichment. See Table A.12
e. Decay heat per assembly	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel Assembly Weight:	≤ 1,680 lbs
i. Operating parameters during irradiation of the assembly	
Average in-core soluble boron concentration	≤ 1000 ppmb
Average core outlet water temperature	≤ 601 K for array/classes 15x15D, E, F and H
	≤ 610 K for array/classes 17x17A, B and C
Average specific power	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F and H
	≤ 61.61 Kw/kg-U for array/classes 17x17A, B and C

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	14x14E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	N/A	N/A
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	~~≤ 5.0~~ *(Note 5)*	~~≤ 5.0~~ *(Note 5)*	~~≤ 5.0~~ *(Note 5)*
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	15x15G	15x15H	16x16A	17x17A	17x17B	17x17C
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	~~≤5.0~~*(Note 5)*	N/A	~~≤5.0~~*(Note 5)*	~~≤5.0~~*(Note 5)*	~~≤5.0~~*(Note 5)*
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

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

2. ZR. Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. *Minimum burnup and maximum initial enrichment as specified in Table A.12.*

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to store only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 5)
BWR FUEL CHARACTERISTICS (Note 1)

Notes:

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

2. ZR designates cladding material made from Zirconium or Zirconium alloys.

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

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

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

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

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

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+."It has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 12	≤ 39,500	≥ 3.2
≥ 15	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % ^{235}U)
≥ 16	≤ 42,000	≥ 3.09
≥ 16	≤ 37,500	≥ 2.6
≥ 16	≤ 30,000	≥ 2.1

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCE COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (Years)
BPRAs	≤ 15,998	≥ 24
TPDs	≤ 118,674	≥ 11
RCCAs	≤ 125,515	≥ 9
Cf neutron source	≤ 15,998	≥ 24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤ 45,361	≥ 19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤ 88,547	≥ 9

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 12	≤ 24,500	≥ 2.3
≥ 14	≤ 29,500	≥ 2.6
≥ 16	≤ 34,500	≥ 2.9
≥ 19	≤ 39,500	≥ 3.2
≥ 20	≤ 42,500	≥ 3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 8	≤ 24,500	≥ 2.3
≥ 9	≤ 29,500	≥ 2.6
≥ 12	≤ 34,500	≥ 2.9
≥ 14	≤ 39,500	≥ 3.2
≥ 19	≤ 44,500	≥ 3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

FUEL ASSEMBLY ARRAY/CLASS	Configuration (Note 2)	Maximum Enrichment (wt% ^{235}U)	MINIMUM BURNUP (B) AS A FUNCTION OF INITIAL ENRICHMENT (E) (Note 1) (GWD/MTU)
15X15D, E, F, H	A	4.79	B = +(1.1483) * E^3 -(13.4246) * E^2 +(63.2842) * E - 71.4084
	B	4.54	B = +(1.535) * E^3 -(16.895) * E^2 +(73.48) * E - 79.05
	C	4.64	B = +(1.23) * E^3 -(14.015) * E^2 +(64.365) * E - 69.9
	D	4.59	B = +(1.34) * E^3 -(15.13) * E^2 +(68.24) * E - 74.07
17x17A, B, C	A	4.70	B = +(0.74) * E^3 -(8.749) * E^2 +(47.7133) * E - 57.8113
	B	4.31	B = +(1.1767) * E^3 -(12.825) * E^2 +(60.7983) * E - 67.83
	C	4.45	B = +(1.3633) * E^3 -(14.815) * E^2 +(66.5517) * E - 73.07
	D	4.38	B = +(1.32) * E^3 -(14.5) * E^2 +(66.39) * E - 73.56

NOTES:

1. *E = Initial enrichment, i.e., for 4.05 wt.%, E = 4.05.*
2. *See Table A.13.*

Table A.13

LOADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• *Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or* • *Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.*
B	• *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank.* • *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.*
C	• *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWd/mtU of the assembly.* • *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.*
D	• *Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWd/mtU of the assembly.* • *The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.*

REFERENCE:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12 dated *TBD*.

HOLTEC INTERNATIONAL

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Non-Proprietary Attachment 4

Updated Revised Certificate of Compliance

(50 pages plus this cover sheet)

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 4

MODEL NO. HI-STAR 100 SYSTEM

NRC FORM 618
(3-96)
10cfr71

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIALS PACKAGES

1.a CERTIFICATE NUMBER	b. REVISION NUMBER	c. PACKAGE IDENTIFICATION NUMBER	d. PAGE NUMBER	e. TOTAL NUMBER PAGES
9261	4	USA/9261/B(U)F-85	1	11

2. PREAMBLE

a. This certificate is issued to certify that the packaging and contents described in Item 5 below, meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of and country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO (*Name and Address*)	b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:
Holtec International Holtec Center 555 Lincoln Drive West Marlton, NJ 08053	Holtec International application dated October 23, 1995, as supplemented c. DOCKET NUMBER 71-9261

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10CFR Part 71, as applicable, and the conditions specified below.

5.

5.a. Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses for the package are called out in drawings listed below.

Multi-Purpose Canister

There are six Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions, except those MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design. A single overpack design is provided that is capable of containing each type of MPC. The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designe· to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is

5. a. (2) Description (continued)

designed to contain up to 32 PWR fuel assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. Any MPC-68 loaded with material classified as fuel debris is designated as MPC-68F. Any MPC-24E loaded with material classified as fuel debris is designated as MPC-24EF.

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring. The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design. The fuel basket designs vary based on the MPC model. For the HI-STAR 100 System transporting fuel debris in a MPC-68F or MPC-24EF, the MPC provides the second inner container, in accordance with 10CFR71.63. The MPC pressure boundary is a strength-welded enclosure constructed entirely of a stainless steel alloy.

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12:

(a) HI-STAR 100 MPC-24 Fuel Basket	Drawing 3926, Sheets 1-4, Rev. 5
(b) HI-STAR 100 MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 4

5. a. (3) Drawings (continued)

(c) HI-STAR 100 MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev. 4
(d) HI-STAR 100 MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev. 8
(e) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 5
(f) HI-STAR 100 Impact Limiters	Drawing C1765, Sheets 1-6, Rev. 1; and Sheet 7, Rev. 0
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 1
(h) Trojan MPC Spacer	Drawing 4111, Sheets 1 and 2, Rev. 0
(i) Trojan Failed Fuel Can	SNC Drawings PFFC-001, Rev. 8 and PFFC-002, Sheets 1 and 2, Rev. 7
(j) Trojan Failed Fuel Can Spacer	Drawing 4122, Sheets 1 and 2, Rev. 0
(k) Holtec Damaged Fuel Container for Trojan plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(l) HI-STAR 100 MPC-32 Fuel Basket	Drawing 3927, Sheets 1-4, Rev. 6

5. b. Contents

(1) Type and Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(g) below are authorized for transportation.

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

5.b.1.(b) Definitions (continued)

Damaged Fuel Containers (or Canisters) (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10 and 1.2.11 of the HI-STAR 100 System SAR, Rev. 12.

Fuel Debris is ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 0.

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

5.b.1.(b) Definitions (continued)

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR 100 System SAR, Rev. 12.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. 12.

ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the two limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPC-68s and MPC-68Fs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the damaged fuel assemblies or the intact fuel assemblies.

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining Zircaloy-clad intact fuel assemblies in the MPC shall meet the more restrictive of the two limits for the 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies or the applicable Zircaloy-clad fuel assemblies.

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For the MPC-32, the administrative procedures to ensure that the cask will be loaded with fuel that is within the specifications should include a measurement that confirms the reactor record for each assembly. Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedure in comparison to procedures that measure each assembly.

The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For confirmation of assembly reactor burnup record(s), the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by a combination of the uncertainties in the record value and the measurement.

c. Transport Index for Criticality Control

The minimum transport index to be shown on the label for nuclear criticality control: 0

6. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

   a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the following provisions:

6.a (continued)

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b above.

(2) Before each shipment, the licensee or shipper shall verify and document that each of the requirements of 10 CFR 71.87 has been satisfied.

(3) The package must satisfy the following leak testing requirements:

(a) All overpack containment boundary seals shall be leak tested to show a total leak rate of not greater than 4.3×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} atm cm^3/sec (helium) and shall be performed:

(i) within the 12-month period prior to each shipment;
(ii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iii) After each seal replacement.

(b) Within 30 days before each shipment, all overpack containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} atm cm^3/sec. If leakage is detected on a seal, then the seal must be replaced and leak tested per Condition 6.a(3)(a) above.

(c) Each overpack containment boundary seal must be replaced after each use of the seal.

(4) The relief devices on the neutron shield vessel shall be replaced every 5 years.

(5) MPC-68F and MPC-24EF shall be leak tested prior to shipment to show a leak rate of no greater than 5×10^{-6} atm cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} atm cm^3/sec (helium).

(6) MPCs deployed at an ISFSI under 10 CFR 72 prior to transportation may be dried using the vacuum drying method or the Forced Helium Dehydration (FHD) method. MPCs placed directly into transportation service under 10 CFR 71 without first being deployed at an ISFSI must be dried using the FHD method. Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

6.a.(6) (continued)

For those MPCs vacuum dried:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The MPC cavity shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

For those MPCs dried using the FHD System:

(a) Following bulk moisture removal, the temperature of the gas exiting the demoisturizer shall be $\leq$ 21°F for $\geq$ 30 minutes.

(7) Following drying, the MPC shall be backfilled with 99.995% minimum purity helium: > 0 psig and $\leq$ 44.8 psig at a reference temperature of 70°F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The overpack annulus shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack annulus shall hold a stable pressure of less than or equal to 3 torr for at least 30 minutes.

(9) Following vacuum drying, the overpack shall be backfilled with helium to $\geq$ 10 psig and $\leq$ 14 psig.

(10) The following fasteners shall be tightened to the torque values specified below:

Fastener	Torque (ft-lbs)
Overpack Closure Plate Bolts	2985 $\pm$ 90
Overpack Vent and Drain Port Plugs	45 +5/-2
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0

(11) Verify that the appropriate fuel spacers, as necessary, are used to position the fuel in the MPC cavity.

b. All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed and shall include the following provisions:

(1) The overpack lifting trunnions shall be tested at 300% of the maximum design lifting load.

6.b (continued)

(2) The MPC shall be pressure tested in accordance with ASME Section III, Subsection NB, Article NB-6110 and applicable sub-articles. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of the design pressure. The minimum hydrostastic test pressure shall be 125 psig. If pneumatic testing is used, the MPC shall be pressure tested to 120% of the design pressure. The minimum pneumatic test pressure shall be 120 psig.

(3) The overpack shall be pressure tested to 150% of the Maximum Normal Operating Pressure (MNOP). The minimum test pressure shall be 150 psig.

(4) The MPC lid-to-shell (LTS) weld shall be verified by either volumetric examination using the Ultrasonic (UT) method or multi-layer liquid penetrant (PT) examination. The root and final weld layers shall be PT examined in either case. If PT alone is used, additional intermediate PT examination(s) shall be conducted after each approximately 3/8 inch of the weld is completed. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PV Section III, NB-5350. The inspection results, including all relevant indications shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(5) The radial neutron shield shall have a minimum thickness of 4.3 inches and the impact limiter neutron shields shall have a minimum thickness of 2.5 inches. Before first use, the neutron shielding integrity shall be confirmed through a combination of fabrication process control and radiation measurements with either loaded contents or a check source. Measurements shall be performed over the entire exterior surface of the radial neutron shield and each impact limiter using, at a maximum, a 6 x 6 inch test grid.

(6) Periodic verification of the neutron shield integrity shall be performed within 5 years of each shipment. The periodic verification shall be performed by radiation measurements with either loaded contents or a check source. Measurements shall be taken at three cross sectional planes through the radial shield and at four points along each plane's circumference. The average measurement results from each sectional plane shall be compared to calculated values to assess the continued effectiveness of the neutron shield. The calculated values shall be representative of the loaded contents (i.e., fuel type, enrichment, burnup, cooling time, etc.) or the particular check source used for the measurements.

(7) The first fabricated HI-STAR 100 overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the Design Drawings specified in Section 1.4 of the SAR, Rev. 9. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures. The test must demonstrate that the overpack is fabricated adequately to meet the design heat transfer capability.

6.b (continued)

(8) For each package, a periodic thermal performance test shall be performed every 5 years or prior to next use, if the package has not been used for transport for greater than 5 years, to demonstrate that the thermal capabilities of the cask remain within its design basis.

(9) The MPC neutron absorber's minimum acceptable ^{10}B loading is 0.0267 g/cm^2 for the MPC-24 and 0.0372 g/cm^2 for the MPC-24E, MPC-24EF, MPC-32, and MPC-68; and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques. The neutron absorber test requirements in Section 8.1.5.3 of the HI-STAR 100 SAR are hereby incorporated by reference into this CoC.

(10) a. The minimum flux trap size for the MPC-24 is 1.09 inches.

b. The minimum flux trap sizes for the generic MPC-24E and MPC-24EF are 0.776 inch for cells 3, 6, 9,and 22; and 1.076 inch for the remaining cells.

c. The minimum flux trap sizes for the Trojan MPC-24E and MPC-24EF are 0.526 inch for cells 3, 6, 9,and 22; and 1.076 inch for the remaining cells.

(11) a. The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

b. The minimum fuel cell pitch for the MPC-32 is 9.158 inches.

(12) The package containment verification leak test shall be per ANSI N14.5-1997.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

9. The personnel barrier shall be installed at all times while transporting a loaded overpack.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

11. Expiration Date: TBD

Attachment: Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, dated TBD.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: TBD

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Table A.1 (Page 1 of 20)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly	
i. ZR Clad:	≤ 833 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

Table A.1 (Page 2 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies listed in Table A.3, with or without Zircaloy channels, and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class.
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR Clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt% ^{235}U.
ii. SS Clad:	An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% ^{235}U.
e. Decay heat per assembly	
i. ZR Clad:	≤ 272 Watts, except for array/class 8x8F fuel assemblies, which shall have a decay heat ≤ 183.5 Watts
ii. SS Clad:	≤ 83 Watts
f. Fuel assembly length:	≤ 176.2 inches (nominal design)
g. Fuel assembly width:	≤ 5.85 inches (nominal design)
h. Fuel assembly weight	≤ 700 lbs, including channels

Table A.1 (Page 3 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 4 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum Planar-Average Initial Enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table A.1 (Page 5 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1(Page 6 of 20)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods Per Thoria Rod Canister:	≤ 18
d. Decay heat per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	≤ 27 kg/canister
g. Fuel cladding O.D.:	≥ 0.412 inches
h. Fuel cladding I.D.:	≤ 0.362 inches
i. Fuel Pellet O.D.:	≤ 0.358 inches
j. Active fuel length:	≤ 111 inches
k. Canister weight:	≤ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1 (Page 7 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)
g. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table A.1 (Page 8 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.8 wt% ^{235}U.
e. Fuel assembly length:	≤ 135.0 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight	≤ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 9 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 10 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide(MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 400 lbs, including channels

Table A.1 (Page 11 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table A.1 (Page 12 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight	$\leq$ 550 lbs, including channels and damaged fuel container

Table *A.1(Page 13 of* 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	An assembly post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

Table A.1(Page 14 of 20)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC:

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris. The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR intact fuel assemblies;
2. MOX BWR intact fuel assemblies;
3. Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers;
4. MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

Table A.1(Page 15 of 20)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR Clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv. Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup ss specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat $\leq$ 833 Watts. Trojan plant fuel decay heat: $\leq$ 725 Watts
ii. SS Clad:	$\leq$ 488 Watts
e. Fuel assembly length:	$\leq$ 176.8 inches (nominal design)
f. Fuel assembly width:	$\leq$ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	$\leq$ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1(Page 16 of 20)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan plant fuel.

F. One (1) Trojan plant Sb-Be and/or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1(Page 17 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR or Stainless Steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. Zr Clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS Clad:	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv. Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup ss specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

Table A.1(Page 18 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

2. Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

Table A.1(Page 19 of 20)
Fuel Assembly Limits

V. MPC MODEL: MPC-24EF

A. Allowable Contents (continued)

3. Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.11
d. Fuel assembly length:	$\leq$ 169.3 inches (nominal design)
e. Fuel assembly width:	$\leq$ 8.43 inches (nominal design)
f. Fuel Assembly Weight:	$\leq$ 1,680 lbs, including non-fuel hardware and DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be transported in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan plant fuel.

F. One (1) Sb-Be and/or two (2) Cf neutron sources may be transported. Each neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly or fuel classified as fuel debris.

Table A.1(Page 20 of 20)
Fuel Assembly Limits

VI. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications:

a. Cladding type:	ZR
b. Maximum Initial Enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, maximum and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly	Calculated value as a function of initial enrichment. See Table A.12
e. Decay heat per assembly	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel Assembly Weight:	≤ 1,680 lbs
i. Operating parameters during irradiation of the assembly	
Average in-core soluble boron concentration	≤ 1000 ppmb
Average core outlet water temperature	≤ 601 K for array/classes 15x15D, E, F and H
	≤ 610 K for array/classes 17x17A, B and C
Average specific power	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F and H
	≤ 61.61 Kw/kg-U for array/classes 17x17A, B and C

B. Quantity per MPC: Up to 32 PWR intact fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-32.

E. Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	14x14E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	N/A	N/A
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15A	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	15x15G	15x15H	16x16A	17x17A	17x17B	17x17C
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

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

2. ZR. Designates cladding material made of Zirconium or Zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to store only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10A	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 5)
BWR FUEL CHARACTERISTICS (Note 1)

Notes:

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

2. ZR designates cladding material made from Zirconium or Zirconium alloys.

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

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

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

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

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

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+."It has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

Table A.4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 12	≤ 39,500	≥ 3.2
≥ 15	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT LIMITS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % ^{235}U)
≥ 16	≤ 42,000	≥ 3.09
≥ 16	≤ 37,500	≥ 2.6
≥ 16	≤ 30,000	≥ 2.1

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCE COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-irradiation Cooling Time (Years)
BPRAs	≤ 15,998	≥ 24
TPDs	≤ 118,674	≥ 11
RCCAs	≤ 125,515	≥ 9
Cf neutron source	≤ 15,998	≥ 24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤ 45,361	≥ 19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤ 88,547	≥ 9

Table A.10

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
$\geq$ 12	$\leq$ 24,500	$\geq$ 2.3
$\geq$ 14	$\leq$ 29,500	$\geq$ 2.6
$\geq$ 16	$\leq$ 34,500	$\geq$ 2.9
$\geq$ 19	$\leq$ 39,500	$\geq$ 3.2
$\geq$ 20	$\leq$ 42,500	$\geq$ 3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)
$\geq$ 8	$\leq$ 24,500	$\geq$ 2.3
$\geq$ 9	$\leq$ 29,500	$\geq$ 2.6
$\geq$ 12	$\leq$ 34,500	$\geq$ 2.9
$\geq$ 14	$\leq$ 39,500	$\geq$ 3.2
$\geq$ 19	$\leq$ 44,500	$\geq$ 3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

FUEL ASSEMBLY ARRAY/CLASS	Configuration (Note 2)	Maximum Enrichment (wt% ^{235}U)	MINIMUM BURNUP (B) AS A FUNCTION OF INITIAL ENRICHMENT (E) (Note 1) (GWD/MTU)
15X15D, E, F, H	A	4.79	B = +(1.1483) * E^3 -(13.4246) * E^2 +(63.2842) * E - 71.4084
	B	4.54	B = +(1.535) * E^3 -(16.895) * E^2 +(73.48) * E - 79.05
	C	4.64	B = +(1.23) * E^3 -(14.015) * E^2 +(64.365) * E - 69.9
	D	4.59	B = +(1.34) * E^3 -(15.13) * E^2 +(68.24) * E - 74.07
17x17A, B, C	A	4.70	B = +(0.74) * E^3 -(8.749) * E^2 +(47.7133) * E - 57.8113
	B	4.31	B = +(1.1767) * E^3 -(12.825) * E^2 +(60.7983) * E - 67.83
	C	4.45	B = +(1.3633) * E^3 -(14.815) * E^2 +(66.5517) * E - 73.07
	D	4.38	B = +(1.32) * E^3 -(14.5) * E^2 +(66.39) * E - 73.56

NOTES:

1. E = Initial enrichment, i.e., for 4.05 wt.%, E = 4.05.
2. See Table A.13.

Table A.13

LOADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank; that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWd/mtU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWd/mtU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCE:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12 dated TBD.

HOLTEC INTERNATIONAL

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U. S. Nuclear Regulatory Commission
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Document ID 5014573

Non-Proprietary Attachment 5

Affidavit Pursuant to 10 CFR 2.390

(5 pages plus this cover sheet)

U.S. Nuclear Regulatory Commission
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Document ID 5014573

AFFIDAVIT PURSUANT TO 10CFR2.390

I, Evan Rosenbaum, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is two proprietary attachments to Holtec letter Document ID 5014573 containing Holtec responses to RAIs, labeled by the NRC as proprietary, in support of LAR 9261-3 for our HI-STAR 100 System Certificate of Compliance (Docket 71-9261).

This information is considered proprietary to Holtec International as is appropriately annotated as such.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

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(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have

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been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10CFR2.390

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

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AFFIDAVIT PURSUANT TO 10CFR2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Mr. Evan Rosenbaum, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 26th day of July, 2005.

Evan Rosenbaum

Evan Rosenbaum
Holtec International

Subscribed and sworn before me this 26th day of July, 2005.

Maria C. Massi

MARIA C. MASSI
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2010