HOLTEC INTERNATIONAL

HI-STAR 100 CERTIFICATE OF COMPLIANCE 71-9261

CERTIFICATE AMENDMENT REQUEST 9261-1

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November 24, 1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: NRC 10 CFR 71 Certificate of Compliance No. 9261, Docket No. 71-9261
License Amendment Request 9261-1

References:
1. Holtec Project No. 5014
2. Holtec Safety Analysis Report No. HI-951251, Revision 8
3. Holtec License Amendment Request 1008-1, dated November 24, 1999

Dear Sir:

Pursuant to 10 CFR 71.31(b), Holtec International hereby submits License Amendment Request (LAR) 9261-1, Revision 0, proposing certain amendments to 10 CFR 71 Certificate of Compliance (CoC) No. 9261 for the HI-STAR 100 transportation package. Information describing and justifying the changes requested by this LAR is contained in the attachments listed below. In preparing this amendment request package, we have intentionally included non-mandatory material, such as marked-up and final versions of the CoC, and proposed Safety Analysis Report (SAR) changes. This non-mandatory information adds to the overall bulk of the submittal, but should greatly facilitate the NRC staff's review effort. A final Revision 9 of the HI-STAR 100 SAR will be submitted within 90 days of the issuance of the amended Certificate of Compliance.

Attachment 1: Summary of Proposed Changes, including the descriptions, reasons, and justifications for the proposed changes.

Attachment 2: Mark-ups of Proposed CoC Changes, Including Appendix A (strikeout/italic form).

Attachment 3: Proposed Revised CoC, Including Appendix A (final form).

Attachment 4: Draft New Certificate Drawings.

Attachment 5: Proposed Revision 9 Changes to the HI-STAR Safety Analysis Report.

This LAR proposes changes to the CoC, including Appendix A; the design drawings referenced by the CoC; and the SAR which include (1) editorial corrections and clarifications, (2) revisions

change: NRC PDR ltr encl 1 1w/0 prop NF01

PDR ADOCK 07109261

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to limits for existing fuel array/classes, (3) two new fuel array/classes, (4) two new fuel canisters, and (5) antimony-beryllium neutron sources. The new fuel canisters added are those in which Dresden Unit 1 fuel assemblies previously stored at West Valley are now stored in Dresden Units 2 and 3 spent fuel pools. Please note that authorization for *transportation* of PWR fuel assemblies containing non-fuel hardware in the HI-STAR 100 System is not being requested at this time. Authorization for *storage* of non-fuel hardware (Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs)) *is* being requested under a separate LAR for the HI-STAR 100 Part 72 CoC 1008 (Reference 3). Aside from this difference, the certificate drawings discussed below, and certain editorial changes that are unique to the respective CoCs, the amendment requests for HI-STAR storage and transportation are identical.

Also included in this amendment request is a set of "certificate drawings" proposed to be added as Appendix B to the CoC in accord with NRC's previous instructions. These certificate drawings contain all dimensional, welding, NDE, and Code compliance information appropriate for a transport CoC, but dispense with the non-significant bits of data which have no reference in the SAR, but which in the real world of custom fabrication would, inevitably, vary from one overpack unit to another. Recognizing the need to pre-empt repetitive amendment requests, the SFPO had asked us prior to the certification issuance to recast our existing licensing drawings on the SAR into Certificate Drawings with all safety-significant information clearly set down. Unfortunately, because of paucity of time, the review of these enhanced drawings could not occur before the issuance of the transport CoC. Through this amendment request we seek to fulfill our commitment with respect to the Certificate Drawings that could not be achieved in the original CoC. These drawings are fully consistent with licensing drawings submitted with the sister storage amendment request number 1008-1 (Ref. 3). They are numbered the same except that the certificate drawing numbers are prefixed with a "C." Certain of the "C" drawings, however, are unique to transportation (e.g., impact limiter drawings). These are enhanced versions of the drawings contained in Revision 8 of the transportation SAR.

In summary, the certificate drawings, created to ensure a high-quality transportation package meeting all licensing requirements, minimize extraneous details, but contain all critical dimensions. Extraneous dimensions/notes which do not bear on the safety analyses have been deleted to minimize the need for repetitive and/or non-safety significant amendment requests in the future. In all cases the safety margins reported in our SAR and in the NRC's Safety Evaluation Report remain robust. All changes in the drawings and TSAR text material have been subjected to our rigorous multi-disciplinary engineering change acceptance review process and appropriately documented in our quality files.

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In order to be able to implement these enhancements and ensure our ability to certify all existing and future HI-STAR 100 Systems for dual-purpose use, we request approval of this amendment request by February 28, 2000.

This submittal also contains information in the form of a Holtec Standard Procedure (HSP-107) which is commercially sensitive to Holtec International and is treated by us with strict confidentiality. This information is of the type described in 10CFR2.790(b)(4). The entirety of this procedure is considered proprietary to Holtec. The attached affidavit sets forth the bases for which the information is required to be withheld by the NRC from further disclosure, consistent with these considerations and pursuant to the provisions of 10CFR2.790(b)(1). It is therefore requested that the proprietary information enclosed be withheld from public disclosure in accordance with applicable NRC regulations.

We appreciate the SFPO's efforts in working through this first-of-a-kind evolution in dry spent fuel storage and transportation technology.

Sincerely,

Brian Gutherman
Licensing Manager

Approved:

K P Singh

K. P. Singh, Ph.D., P.E.
President and CEO

Document I.D.: 5014355

Attachments: 1 – 5: As Stated Above
6. Affidavit Pursuant to 10 CFR 2.790

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Enclosure: Holtec Standard Procedure HSP-107

Cc: Ms. Virginia Tharpe, USNRC, (10 hard copies, w/attach and encl., and floppy disk of cover letter and Attachments 1 through 3)

Cc: Mr. E. W. Brach (cover letter only)
Ms. S. Frant-Shankman (cover letter only)
Mr. R. Chappell (cover letter only)

Technical Concurrence:

Mr. Bernard Gilligan (Configuration Control)

Dr. Alan Soler (Structural Evaluation) Alan Soler

Dr. Indresh Rampall (Thermal Evaluation) Indresh Rampall

Mr. Kris Cummings (Containment Evaluation)

Dr. Everett Redmond II (Shielding Evaluation)

Dr. Stefan Anton (Criticality Evaluation) for Stefan Anton

Mr. Steve Agace (Operations) S. Agace

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Dr. Seymour Raffety	Dairyland Power
Mr. John Sanchez	Consolidated Edison Company
Ms. Kathy Picciott	Niagara Mohawk Power Corporation
Mr. John Donnell	Private Fuel Storage, LLC (SWEC)

SUMMARY OF PROPOSED CHANGES

SECTION I – PROPOSED CHANGES TO CERTIFICATE OF COMPLIANCE 9261

Proposed Change No. 1

Certificate of Compliance, Section 5.a.(2), "Description", and Item 7 (page 7):

The maximum gross package weight is increased from 280,000 lbs to 282,000 lbs. Note that this change requires a concomitant change to Certificate Drawing C1782 (see Proposed Change Number 2).

Reason for Proposed Change

To correct the package weight in the CoC to match the maximum analyzed weight given in Safety Analysis Report (SAR) Table 2.2.4.

Justification for Proposed Change

The HI-STAR package was analyzed for its initial certification assuming a total package weight of 282,000 lbs, including impact limiters. The current revision of the CoC states 280,000 lbs because there is a discrepancy between the weight value in Revision 8 of the SAR and on the drawing. This proposed change and the proposed drawing change make the CoC, the drawing, and the SAR consistent.

Proposed Change No. 2

Certificate of Compliance, Section 5.a.(3), "Drawings":

a. The drawing list is revised to reflect the replacement of the previous Holtec Design Drawings with revised Holtec Certificate ("C") drawings as the set of drawings which are an integral part of the CoC. These "C" drawings are included in Attachment 4. This change also includes a proposal to add a new Appendix B to the CoC which will contain the "C" drawings and delete the reference to the SAR.

b. The drawing list is revised to add the Transnuclear Dresden Unit 1 (TN/D-1) damaged fuel canister and the D-1 Thoria Rod Canister. These new drawings are included in Attachment 4.

Reason for Proposed Changes

a. The drawings referenced in the current, approved transportation CoC are the same drawings as those used to license the HI-STAR 100 System for storage (CoC 1008). They contain a level of detail suitable for licensing review but too much detail for drawings considered an integral part of the CoC (i.e., which cannot be changed without a certificate amendment). The detailed drawings (as revised under the amendment request for storage CoC 1008) will remain in the HI-STAR SAR for historical purposes, consistent with the balance of the transport SAR material. The "C" drawings, and changes thereto will be controlled with the CoC itself as Appendix B.

b. These drawings are added to reflect the authorization of a Transnuclear D-1 damaged fuel canister and a D-1 Thoria Rod Canister as allowed contents for the HI-STAR 100 packaging. The addition of these canisters to Appendix A of the CoC as authorized package contents is discussed below as a separate change.

Justification for Proposed Changes

a. The reason for the changes is that the storage version of the detailed design drawings may be changed by licensees under the provisions of 10 CFR 72.48. There is no analogous process under Part 71. Therefore, a transportation CoC amendment would be required for any change made under §72.48 which affects a drawing common to transportation and storage in order to certify the affected cask for transportation. This proposed change to more appropriately detailed transportation certificate drawings is designed to eliminate the need for Part 71 CoC amendments for non-safety-significant changes to the design drawings. See Section III for additional information on drawings. This change will reduce the burden on Holtec and NRC staff resources which would be assigned to prepare and review, respectively, amendment requests of little or no safety significance. The provisions of 10 CFR 72.48 for transportation provide the appropriate administrative controls to ensure changes to the detailed drawings which create an unreviewed safety question are reviewed and approved by the NRC before implementation.

b. These drawings are proposed to be included in Appendix B the CoC to ensure only the approved canisters are loaded for transportation in the HI-STAR 100 System. It also ensures any changes to the drawings are controlled through an amendment to the CoC.

Proposed Change No. 3

Certificate of Compliance, Section 5.b.(1).(b), Definitions:

a. The definition of **Damaged Fuel Assemblies** is revised as shown in the attached marked-up pages of the CoC to make the distinction between damaged fuel and fuel debris consistent with that used in storage CoC 1008.

b. The definition of **Damaged Fuel Containers** is revised as shown in the attached marked-up pages of the CoC to include the Holtec-designed DFC and a Transnuclear (TN/D-1) DFC[1] currently containing D-1 fuel. Drawings of the TN/D-1 DFC are contained in Attachment 4 and will be added to new CoC Appendix B and SAR Section 1.4 (see Proposed Change No. 2).

c. The definition of **Fuel Debris** is revised as shown in the attached marked-up pages of the CoC to be consistent with the definition used in HI-STAR storage CoC 1008.

d. The definition of **Planar-Average Initial Enrichment** is revised as shown in the attached marked-up pages of the CoC to delete the word "simple" and add the word "initial" to be consistent with the definition used in HI-STAR storage CoC 1008.

Reason for Proposed Changes

a., c., and d. The current definitions of these terms are inconsistent between the storage and transportation certificates of compliance.

b. There are a significant number of Dresden Unit 1 fuel assemblies meeting the HI-STAR fuel specifications which are currently stored in TN/D-1 damaged fuel containers (DFCs). Authorizing this fuel for transportation in the HI-STAR 100 packaging without having to remove it from the TN/D-1 DFCs and load it into the Holtec DFCs will avoid imposing undue burden on the general licensee with no additional safety benefit.

Justification for Proposed Changes

a., c. and d. Required for editorial consistency.

[1] The terms Damaged Fuel Container and Damaged Fuel Canister are used interchangeably throughout this document and "DFC" is applicable to both.

b. The justification for this proposed change is provided below, arranged by technical discipline, as applicable. Conforming changes to the SAR are summarized in Section II of this attachment and included in Attachment 5.

Structural Evaluation

The TN/D-1 DFC was previously approved for use in the TN-9 transportation package. In addition, the TN/D-1 DFC has been structurally evaluated by Holtec International and found to meet all required design requirements for transportation in the HI-STAR 100 package. The details of this evaluation are contained in proposed new SAR Appendix 2.AO, included in Attachment 5 to this letter. All required safety margins are greater than zero or, in other words, the factors of safety are greater than 1.0.

The SAR Chapter 2 NUREG-1617/10 CFR 71 compliance matrix has been revised to address the new DFCs and the supporting appendix. Since all required text changes are confined to the new appendix, no new chapter text is required.

Thermal Evaluation

Transportation of damaged fuel and fuel debris meeting the specifications of the CoC is permitted in the HI-STAR MPC-68 and –68F when encased in a DFC. The thermal characteristics of the TN/D-1 DFC and the Holtec DFC shown on Design Drawing C1783 were compared in support of this amendment request. The TN/D-1 DFC is a square shaped canister box fabricated from 12 gage stainless steel plates. A bounding thermal calculation has been prepared in support of this amendment to determine the most heat resistive fuel from the Low Heat Emitting (LHE) group of assemblies encased in a DFC. It is noted that in this configuration, interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary renders this as bounding. Both canister designs are evaluated and the one exhibiting lower heat dissipation characteristics was adopted for analysis.

In the LHE group of assemblies, the low decay heat load D-1 fuel (approximately 8 kW) guarantees large thermal margins to permit safe transportation of D-1 fuel in the TN/D-1 DFC. The HI-STAR temperature field for a conservatively bounding heat load case was calculated and is reported in proposed revisions to HI-STAR SAR Chapter 3 at Subsection 3.4.1.1.18. A substantial margin of safety has been demonstrated by the analysis.

Shielding Evaluation

Transport of damaged fuel and fuel debris meeting the specifications of the CoC is permitted in the HI-STAR MPC-68 and MPC-68F when encased in a DFC. Sections 5.4.2 and 5.4.3 of the HI-STAR SAR Revision 8 discuss the post-accident shielding evaluation for the damaged fuel. These sections assume that the damaged fuel assemblies and fuel debris collapse to a height of 80 inches. This dimension was calculated based on the inside dimension of the DFC and the dimensions of the fuel assemblies. Since the Transnuclear DFC has a smaller inside dimension than the Holtec DFC, the analysis in Sections 5.4.2 and 5.4.3 of the HI-STAR SAR is applicable and conservative. In addition, the shielding analysis does not take credit for the DFC container in determining the acceptability of storing the approved damaged fuel and fuel debris. Therefore, the use of the TN/D-1 DFC does not affect the shielding analysis and no changes to the Chapter 5 of the SAR are necessary as a result of this proposed change.

Criticality Evaluation

The TN/D-1 DFC was analyzed with the same set of contents used for the analysis of the Holtec DFC documented in Rev. 8 of the HI-STAR 100 SAR. This set includes 6x6 and 7x7 fuel assemblies with various numbers of rods missing, a collapsed assembly and dispersed fuel powder. The maximum k_{eff} values for both DFCs are listed in SAR Table 6.4.5 (Attachment 5). There is no significant difference in reactivity between the two DFCs. Only for one case (collapsed assembly) the reactivity for the TN/D-1 DFC is increased marginally ($\Delta k = 0.0012$) compared to the Holtec DFC. In all other cases, the reactivity for the TN/D-1 DFC is below the reactivity of the Holtec DFC with the same contents. Therefore, with the TN/D-1 DFC used instead of the Holtec DFC, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$.

Proposed Change No. 4

Certificate of Compliance, Section 6.a

The wording of this item is revised to correct a grammatical error as shown in the attached marked-up pages of the CoC.

Reason and Justification for Proposed Change

Editorial.

Proposed Change No. 5

Certificate of Compliance, Section 6.a.(7):

The MPC helium backfill *density* limit is revised to be a maximum helium backfill *pressure* with acceptance criteria as shown in the attached marked-up CoC pages

Reason for Proposed Change

Density limits for helium backfill of the MPC are necessary when basket internal convection heat transfer is relied upon for safe transportation of the spent nuclear fuel. The HI-STAR licensing basis completely neglects this mode of heat transfer. The existing limits on helium backfill density are, therefore, overly restrictive and a change in favor of a simpler requirement is warranted. The change is designed to relieve the users of an unnecessary burden of confirming helium backfill density within in a very narrow range of acceptance.

Justification for Proposed Change

The proposed change to the MPC helium backfill CoC limits requires the users to backfill the MPC up to a maximum helium pressure. This ensures the presence of helium in the MPC free space. Any positive helium pressure in the MPC (i.e., > 1 atm) is consistent with the governing thermal analyses. The positive helium pressure in the MPC along with the helium pressure of between 10 and 14 psig in the HI-STAR overpack annulus (also required by the CoC) provide reasonable assurance of no air inleakage into the MPC cavity during transport operations. The upper pressure limit protects the MPC from potential overpressure during the hypothetical accident scenario where 100% of the fuel rods are assumed to rupture.

Proposed Change No. 6

Certificate of Compliance, Section 6.a.(10):

a. The double asterisk and associated note from the overpack closure plate bolt torque value are deleted.

b. The torque value for the overpack vent and drain port plugs is increased to 45 +5/-0 ft-lbs.

Reason for Proposed Changes

a. To provide flexibility for the user.

b. To provide the proper amount of compression for the port plug seals.

Justification for Proposed Changes

a. The pattern and number of passes used to torque the overpack closure plate bolts is an unnecessary level of detail to be included in the CoC. While this is the recommended pattern and number of passes in the SAR, the user should have the option of using a different process, provided the required bolt torques are met.

b. The seal manufacturer has recommended increasing the port plug torque to ensure sufficient compression of the seal. The depth of the seal groove machined under the heads of the port plugs ensure the seals seat at the higher torque without over-compression.

Proposed Change No. 7

Certificate of Compliance, Section 6.b.(1):

The requirement to perform trunnion acceptance testing in accordance with ANSI N14.6 is deleted.

Reason for Proposed Changes

To delete an unnecessary reference to an ANSI Standard and make the transportation requirements for trunnion testing consistent with HI-STAR storage (CoC 1008).

Justification for Proposed Changes

SAR Subsection 8.1.2.1 provides the necessary guidance for trunnion acceptance testing and subsequent inspections. Referencing ANSI N14.6 provides unneeded additional requirements (e.g., annual testing and non-destructive examination beyond visual examination).

Proposed Change No. 8

Certificate of Compliance, Section 6.b.(7):

The requirement to perform heat transfer capability testing on every fabricated HI-STAR 100 overpack is revised to require this testing on only the first fabricated overpack.

Reason for Proposed Change

To make the transportation requirements for heat transfer capability testing consistent with HI-STAR storage (CoC 1008).

Justification for Proposed Change

HI-STAR overpack design and fabrication are controlled by the design drawings and the fabrication procedures, including quality control inspections. The overall heat transfer capability of the overpack will be confirmed by testing of the first fabricated overpack as stated in the CoC. Additional testing of subsequent fabricated overpacks would be expected to yield similar results as the first test and is therefore deemed unnecessary.

Proposed Change No. 9

Certificate of Compliance, Section 6.b.(9):

The words "verified by" are added after "shall be."

Reason for and Justification Proposed Change

Editorial Correction.

Proposed Change No. 10

Certificate of Compliance, Item 8 (page 7):

The word "least" is added after "at."

Reason and Justification for Proposed Change

Editorial correction.

Proposed Change No. 11

Certificate of Compliance, Appendix A, Table A.1:

MPC-24, Item I.A.1.c and MPC-68, Item II.A.1.d are revised to add "decay heat" after "burnup" and "initial" before "enrichment."

Reason and Justification for Proposed Changes

The tables to which these items are refer include limits on decay heat. The word "initial" is added to be consistent with the use of the term "initial" with maximum enrichment limits in the same table.

Proposed Change No. 12

Certificate of Compliance, Appendix A, Table A.1, Items II.A.3.d, II.A.4.d, III.A.1.d, III.A.2.d, and III.A.3.d:

The "c" in "Cooling time" is changed to lower case.

Reason for and Justification Proposed Change

Editorial Correction.

Proposed Change No. 13

Certificate of Compliance, Appendix A, Table A.1:

For the MPC –68 and MPC-68F, new Items II.A.5 and III.A.7 are added as shown in the attached marked-up pages of the CoC table to allow transportation of up to one Dresden Unit 1 (D-1) Thoria Rod Canister in these MPC models. Drawings of the D-1 Thoria Rod Canister are provided in Attachment 4 and are added to Section 1.4 of the SAR and Appendix B of the CoC (see Section II of this attachment). Conforming revisions are made to CoC Table A.1, Items II.B and III.B.

Reason for Proposed Change

Dresden Unit 1 needs to place one Thoria Rod Canister into transportation to support plant decommissioning.

Justification for Proposed Change

Structural Evaluation

The Dresden Unit 1 Thoria Rod Canister has been structurally evaluated by Holtec International and found to meet all required design requirements for transportation in the HI-STAR 100 system. The details of this evaluation is contained in proposed new SAR Appendix 2.AO, included in Attachment 5 to this letter. All required safety margins are greater than zero or, in other words, the factors of safety are greater than 1.0.

Thermal Evaluation

The Thoria Rod Canister is designed to hold a maximum of 20 fuel rods arrayed in a 5x4 configuration. Eighteen rods are actually in the canister. 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 (< 16,000 MWD/MTIHM). 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 thus 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. All these considerations provide ample assurance that these fuel rods will be stored in a benign thermal environment and therefore remain protected during transportation.

Shielding Evaluation

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

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years.

Table 5.2.29 of proposed Revision 9 of the HI-STAR SAR describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.30 and 5.2.31 of proposed Revision 9 of the HI-STAR SAR shows the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the Thoria Rod Canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.6 and 5.2.14 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

It is obvious that the neutron spectra from the 6x6 fuel assembly bounds the thoria rod neutron spectra 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-68 filled with 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 only one Thoria Rod Canister is proposed to be authorized for transport in the HI-STAR 100 System clearly demonstrates that the Thoria Rod Canister is acceptable for transport in the MPC-68 or the MPC-68F.

Criticality Evaluation

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in proposed new SAR Figure 6.4.10. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.18. 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 outside diameter is 0.412") and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in SAR 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 $k_{eff} < 0.95$.

Containment Evaluation

The HI-STAR containment analyses have been revised to account for several new isotopes associated with the Thoria Rod Canister. These isotopes (Bi-212, Pb-212, Po-216, Ra-224, Rn-220, Th-228 and U-232) had a negligible effect on the resulting doses because only one Thoria Rod Canister is authorized for loading in an MPC-68 or –68F with 67 other design basis BWR assemblies. Therefore, the thoria rod isotopes are not included in the presentation of the containment analysis inputs or results in the SAR.

Proposed Change No. 14

Certificate of Compliance, Appendix A, Table A.1:

New Items II.D and III.D are added as shown in the attached marked-up CoC pages to authorize Dresden Unit 1 fuel assemblies containing one antimony-beryllium neutron source in the assembly lattice for transportation.

Reason for Proposed Change

Dresden Unit 1 needs to place fuel assemblies containing antimony-beryllium neutron sources into transportation to support plant decommissioning.

Justification for Proposed Change

Structural Evaluation

The fuel assembly parameters used in the design basis structural evaluations are not affected by this change. The neutron sources have no impact on component temperatures or fuel assembly size and weight.

Thermal Evaluation

The substitution of antimony-beryllium sources in a fuel assembly in lieu of heat emitting fuel rods is bounded by the existing thermal analyses, which assume decay heat production from the replaced fuel rods.

Shielding Evaluation

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 of Revision 8 of the HI-STAR SAR. 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 14.1 below, 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.

Table 14.1
Comparison of Neutron Source per inch per second for Design Basis 7x7 fuel and Design Basis Dresden Unit 1 fuel.

Assembly	Active fuel length (inches)	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	6.38E+5	N/A	Table 5.2.13 Rev. 8 HI-STAR SAR 39.5 GWD/MTU and 15 year cooling
6x6 design basis	110	2.85E+5	3.45E+5	Table 5.2.14 Rev. 8 HI-STAR SAR
6x6 design basis MOX	110	3.67E+5	4.27E+5	Table 5.2.17 Rev. 8 HI-STAR SAR

<u>Criticality Evaluation</u>

The reactivity of a fuel assembly containing neutron sources is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a k_{eff} less than 1.0, any neutron population produced 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 also not lead to an increase of reactivity.

<u>Proposed Change No. 15</u>

<u>Certificate of Compliance, Table A.1, Item III.A.1:</u>

The word "specified" is added after the word "criteria."

Reason for and Justification Proposed Change

Editorial Correction.

Proposed Change No. 16

Certificate of Compliance, Appendix A, Tables A.2 and A.3:

The notes at the bottom of all but the last sheet of both tables are deleted. The remaining notes at the end of Tables A.2 and A.3 are revised as described below and as shown in the attached marked-up pages of the CoC. Pointers to these notes are also revised accordingly.

a. Note 1 in Table A.3 is revised to reduce the scope of the note to apply to dimensional limits only.

b. Note 3 in both tables is revised to clarify the intent. Previous Notes 3 through 9 in Table A.3 are now Notes 4 through 10.

c. Re-numbered Note 4 in Table A.3 is revised to increase the allowable weight percent of U-235 in the MOX rods of fuel assembly array/class 6x6B from 0.612 to 0.635. This note is also clarified to state that the weight percentages are to be calculated based on the total fuel weight (i.e., uranium oxide plus plutonium oxide).

d. New Note 11 is added to Table A.3.

e. New Note 12 is added to Table A.3.

f. New Note 13 is added to Table A.3.

Reason for Proposed Changes

a. The current Note 1 in Table A.3 addresses both dimensional limits and uranium weight limits. This was inappropriate since the note conveys different information for these different data types. Splitting the note removes a potential source of confusion for the user and makes HI-STAR transportation consistent with HI-STAR storage (CoC 1008) and HI-STORM (Docket 72-1014).

b. As currently worded, it is unclear whether implementation of the tolerance offered by Note 3 requires adjusting the value of the as-delivered uranium mass for a fuel assembly, or adjusting the uranium mass limit specified in the table for comparison against users' fuel records. The intent is to adjust the uranium mass limit up, as necessary, for comparison against users fuel records. This eliminates a potential poor practice of users adjusting uranium mass values found on fuel records.

c. User feedback indicates that there are fuel assemblies with MOX rods containing less than 1.578 weight percent fissile plutonium in natural uranium. To bound this situation, the uranium content is increased slightly. The second change to Note 4 is proposed to improve clarity regarding the intent of the note.

d. New Note 11 is proposed in response to user feedback that some assemblies may include non-fuel rods which are filled with non-fissile material in lieu of water.

e. New Note 12 is proposed to be added for information on this new array/class.

f. New Note 13 is proposed to address a situation for the 9x9E fuel assembly array/class where one assembly type in the class (SPC 9x9-5) contains rods of different dimensions within the array.

Justification for Proposed Changes

a. Enhances user implementation.

b. None. The tolerance in the mass limit allowed by this note is in the current, approved CoC.

c. All criticality calculations for the 6x6B fuel assembly array/class were re-performed (see proposed revised SAR Table 6.2.38 in Attachment 5). The change in reactivity for this change is small (less than 2σ). This demonstrates that the maximum k_{eff} remains below 0.95 with the increased uranium concentration. The second change is proposed for clarity.

d. Replacing water with a non-fissile material will reduce the amount of moderator without increasing the amount of fissile material. This results in a decreased reactivity. This situation is comparable to the overall reduction of water density analyzed in Section 6.4.2.1 of the SAR, which shows a decrease of reactivity with decreasing water density (i.e. decreasing the amount of water in the cask). The existing calculations assuming water in the water rods are therefore bounding for rods with non-fissile material in lieu of water.

e. New fuel assembly array/class 8x8F represents a unique fuel assembly type known as the QUAD+. New Note 12 is proposed to describe the unique water rod features of this assembly.

f. The SPC 9x9-5 fuel type is configured with two types of fuel rods having differing dimensions. Accordingly, the criticality analyses have been performed considering the varying fuel rod dimensions in the SPC 9x9-5 fuel type. Bounding all fuel rods in the assembly with one set of rod dimensions is not feasible because of excessive dimensional overlap.

<u>Proposed Change No. 17</u>

<u>Certificate of Compliance, Appendix A, Tables A.2 and A.3:</u>

The maximum allowed design initial uranium masses for selected fuel assemblies is increased as shown in the marked-up CoC tables. This affects PWR fuel assembly array/classes 14x14A, 14x14B, 14x14C, 15x15A, 16x16A, 17x17A, 17x17B, and 17x17C in Table A.2 and BWR fuel assembly array/classes 6x6A, 6x6B, 6x6C, 8x8E, 9x9A, 9x9B, 9x9C, 9x9D, 9x9E, 9x9F, 10x10A, 10x10B, and 10x10C in Table A.3.

Reason for Proposed Changes

To respond to user feedback describing certain fuel assemblies which have uranium masses slightly above the specified limit (including the tolerance allowed by Note 3) for the applicable fuel assembly array/class. These changes are required to ensure users can load all of the fuel they plan to place into transportation.

Justification for Proposed Changes

<u>Structural Evaluation</u>

There is no effect on the existing structural evaluation. The increased uranium masses do not cause an increase in the overall assembly weight limits in the same tables of the CoC. These weights (or greater) were used in the structural evaluation. Since the allowed assembly weights are not being changed, the structural evaluation is unaffected.

Thermal Evaluation

There is no effect on the existing thermal evaluation. This is because the allowed heat load for the cask is computed based on the heat transfer characteristics of the cask system and peak cladding temperature limitations. The increase in uranium mass does not impact any assumption made in determining the heat transfer characteristics of the cask system.

Shielding Evaluation

The uranium mass limit is a value that is determined from the shielding analysis. An increase in the mass of uranium will result in an increase in the neutron and gamma source term and decay heat load for a specified burnup and cooling time. The current CoC developed from the analyses in Revision 8 of the HI-STAR SAR provides some margin between the analyzed mass of uranium and the approved mass of uranium as listed in the CoC. The allowable burnup and cooling times in the CoC were developed by a comparison of the calculated decay heat for the design basis assemblies to the allowable decay heat as determined in the thermal analysis and a comparison of the calculated dose rates to the allowable dose rates. The decay heat values that are compared against the limits were calculated using the mass of uranium listed in Chapter 5 of the HI-STAR SAR for the design basis fuel assemblies. Since a lower mass of uranium will result in a lower decay heat, it is conservative, and provides margin, to specify the allowable mass of uranium in the current CoC for the design basis fuel assemblies (B&W 15x15 and 7x7) lower than the values analyzed in Chapter 5.

As discussed in Section 5.2.5 of the HI-STAR SAR Revision 8, the design basis assembly was chosen by comparing the source terms for many different types of assemblies. All of the assemblies were shown to have a lower source term than the design basis fuel assemblies. For additional conservatism the mass of uranium specified in the CoC of Revision 8 of the HI-STAR SAR for these non-design basis fuel assemblies was also specified lower than the mass used in the comparison in Chapter 5. This level of conservatism was unnecessary since the decay heat load and radiation source used to determine the allowable burnup and cooling times for all assemblies was the decay heat load and radiation source from the design basis fuel assemblies. Therefore, there was already a significant margin of conservatism for the non-design basis fuel assemblies by using the design basis fuel to determine the allowable burnup and cooling times. Section 5.2.5.3 of Revision 8 of the HI-STAR SAR provides an indication of the level of conservatism associated with using the design basis decay heat for the non-design basis fuel assemblies.

The proposed change in the CoC is to increase the mass of uranium for the non-design basis fuel assemblies up to the value that was used in the analysis in Chapter 5 of the HI-STAR SAR to determine the design basis fuel assembly. As mentioned above, this change eliminates unnecessary over-conservatism while still maintaining a significant degree of conservatism and margin for the non-design basis fuel assemblies. The allowable mass loading for the design basis fuel assemblies remains unchanged. Therefore, the proposed change does not affect the shielding analysis presented in Revision 8 of the HI-STAR SAR. Additional clarification has been added to the proposed Revision 9 of the HI-STAR SAR to discuss this issue.

Criticality Evaluation

The criticality analyses are not affected by these changes. Criticality analyses for the bounding assembly in each array/class are performed with a fuel (uranium) mass exceeding the mass limit in the CoC for that array/class. This is due to the assumed fuel density of 96.0% of the theoretical fuel density of 10.96 g/cm^3 for all criticality calculations.

Containment Evaluation

As described in the shielding evaluation, the values of uranium mass used in the shielding analyses have not changed. These proposed changes simply increase the allowed uranium masses to those used in the analysis. The source terms used in the containment analyses were taken from the source terms used in the shielding analyses. Therefore, the existing containment evaluation is still bounding for the proposed new uranium mass limits.

Proposed Change No. 18

Certificate of Compliance, Appendix A, Tables A.2 and A.3:

Certain fuel assembly parameter limits are revised as shown in the attached marked-up CoC tables. This affects PWR fuel assembly array/class 14x14C in Table A.2 and BWR fuel assembly array/classes 6x6A, 6x6B, 7x7A, 7x7B, 8x8A, 8x8B, 8x8D, 9x9B, 9x9D, 9x9E, 9x9F, and 10x10C in Table A.3.

Reason for Proposed Changes

To respond to user feedback describing certain fuel assemblies which have parameters outside of the limits in the existing CoC Tables. These changes are required to ensure users can load all of the fuel they plan to transport.

Justification for Proposed Changes

Structural Evaluation

The proposed changes to parameter limits for some of the existing fuel assembly array/classes have no impact on the structural evaluation because the design basis weights used in the analyses (and provided as limits elsewhere in the CoC) are not changed, the design basis temperatures are not changed, and the lengths and widths of the fuel assemblies (also limited by the CoC) are not changed.

Thermal Evaluation

The active fuel length for array/classes 6x6A and 6x6B is proposed to be increased to 120 inches to bound an earlier variant of Dresden-1 fuel. Among the fuel assemblies included in the 6x6A array/class, one particular fuel type was determined to be fabricated with a thinner cladding (0.026 in.) relative to other fuel in this class (minimum 0.030 in. cladding). In the 7x7A array/class of fuel assemblies, minor adjustments to the fuel parameters[2] was necessary to bound Humboldt Bay fuel. Changes to the 7x7B and 8x8B fuel assembly array/classes were necessary to bound the fuel types at Oyster Creek plant. Accordingly, the thermal analysis for these fuel types was evaluated in support of this amendment and additional analyses performed, as required.

A review of the Oyster Creek fuel parameters against the fuel parameters of other fuel types in the same array/classes has revealed no significant differences. The Oyster Creek 7x7 fuel rod mechanical parameters are identical to an existing member of the 7x7B class. The relatively larger pellet diameter (from 0.491 vs. 0.488 in) necessitates an adjustment to the uranium mass limit for this array/class. The Oyster Creek 8x8 fuel rod diameter is slightly larger than other members in the 8x8B class and has a thicker cladding.

An 8x8 fuel assembly used at Browns Ferry and a 9x9 fuel assembly from Grand Gulf, have been evaluated in support of this amendment request to modify the BWR fuel parameters. Likewise, a Millstone Unit 2 14x14 fuel assembly has been evaluated to support modification of the PWR fuel tables. As explained below, these PWR and other BWR fuel have been evaluated in accordance with the NRC approved HI-STAR thermal analysis methodologies to confirm that the HI-STAR 100 temperature field is bounded by the design basis analyses.

The overall HI-STAR thermal analysis methodology is partitioned into two evaluations. The first evaluation pertains to determining the appropriate peak

[2] Cladding thickness change from 0.033 in to 0.0328 in and active fuel length from 79 in to 80 in.

cladding temperature limits for transportation of each proposed fuel type. In the second evaluation, the temperature field in the HI-STAR 100 cask is computed and the resulting cladding temperatures demonstrated to be below the respective temperature limits. The analytical evaluations are further sub-divided in two groups of fuel assemblies classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE fuel assemblies are characterized by low burnup, long cooling time and short active fuel lengths. Consequently, their heat loads are dwarfed by the full active length DB fuel assemblies. The additional Dresden-1 and Humboldt Bay fuel assemblies in the 6x6A and 7x7A array/classes belong to the LHE group of fuel, while the additional Oyster Creek, Browns Ferry, and Grand Gulf fuel assemblies are included in the DB group.

In accordance with the PNL-6189 methodology, peak fuel cladding temperature limits are specified as a function of cladding stress and age of fuel. The cladding stress calculations for the additional fuel are documented in proposed revised SAR Tables 3.3.5 and 3.3.6 in Attachment 5 to this letter. The cladding stress in the additional DB fuel types is bounded by the limiting cladding stress computed previously. An adjustment to the SVEA-96 fuel parameters (an O.D. change by 0.001 inch) is insignificant for the cladding stress evaluation as it is bounded by the design basis cladding stress. Consequently, the age-dependent peak fuel cladding temperature limits do not require changes to accommodate the additional fuel. For the LHE fuel group, the thin-clad Dresden-1 fuel type is determined to be the limiting fuel resulting in a downward shift in the applicable fuel cladding temperature limit. The revised temperature limits for LHE and DB fuel are summarized in proposed revised SAR Table 3.3.7.

The second evaluation pertaining to computation of the HI-STAR 100 cask temperature field is functionally dependent upon the effective conductivity of fuel assemblies loaded in the MPC-68 fuel cells. The LHE fuel assemblies are further analyzed under the assumption that they are loaded while encased in stainless steel DFCs. Due to interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary, this configuration is bounding for the thermal evaluation. Two DFC designs are evaluated – a previously approved Holtec design (Holtec Drawing No. C1783) and an existing TN/D-1 DFC in which some of the Dresden-1 fuel is currently stored (see Proposed Change Number 3.b) . The most resistive fuel assembly determined by analytical evaluation is considered for the HI-STAR 100 cask thermal evaluation. The results of the evaluation of additional fuel types performed in support of this amendment request are summarized in proposed revised Table 3.4.6 for LHE and DB fuel (see Attachment 5).

In both groups investigated, the thermal conductivity of the additional fuels are bounded by the limiting fuel types in each group. For the DB group of fuel assemblies, it is shown that the peak cladding temperature limits for the limiting fuel type adequately cover the additional fuel. The most resistive fuel characteristics also bound the additional fuel in the list of DB fuel types authorized for transportation in the HI-STAR 100 System. Thus, the design basis thermal analysis results envelope the HI-STAR 100 System thermal response when loaded with the additional BWR and PWR fuel. No additional analysis is required. For the LHE group of assemblies, the low decay heat load burden on the HI-STAR 100 cask (~ 8kW) guarantees large thermal margins to permit safe transportation of Dresden-1 and Humboldt Bay fuel. Nevertheless, a conservative analysis was performed and is described in the proposed Revision 9 SAR and the temperature field determined and reported Subsection 3.4.1.1.18 (see Attachment 5).

Shielding Evaluation

The accuracy of the shielding analysis is dependent upon the calculation of the radiation source term. The source term is dependent on the mass of uranium in the fuel assembly. For a specified burnup and cooling time, the radiation source term will increase as the mass of uranium increases. Minor variations in the dimensions of a fuel assembly will have a negligible impact on the radiation source term if the mass of uranium remains constant. The proposed changes to the CoC for the HI-STAR 100 are minor changes to some of the fuel assembly dimensions. Since the allowable uranium mass loadings are not being changed as a result of these change in dimensions it is concluded that these changes will have a negligible effect on the shielding analysis and therefore are not explicitly considered in Revision 9 of Chapter 5 of the HI-STAR SAR.

Criticality Evaluation

For the criticality evaluation, the fuel assemblies are grouped into assembly array/classes. The proposed CoC modifications to fuel assemblies already included are reflected in proposed revised SAR Table 6.2.1 (see Attachment 5). For each assembly array/class, a theoretical bounding assembly is defined. The characteristics of the bounding assembly for each affected array/class was amended to reflect the additional fuel types within an array/class.

Criticality safety calculations were performed for the changed fuel types and the bounding assembly in each array/class to account for the modified dimensions. Table 18.1 below shows the comparison between the maximum k_{eff} for each of the affected array/classes and the corresponding current values (i.e. SAR Rev. 8). The SAR table number containing the detailed results is also listed. The comparison demonstrates that the maximum k_{eff} of each affected class only changes slightly as a result of the changes in fuel assembly characteristics. Furthermore, the highest reactivity calculated for any BWR or PWR class (0.9457 for the bounding assembly in the BWR 10x10A class and 0.9478 for PWR assembly class 15x15F remains unaltered (see proposed Revision 9 SAR Tables 6.2.30 and 6.2.13, respectively). Therefore, with the proposed changes, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all authorized fuel assembly array/classes.

Table 18.1
Comparison of Maximum k_{eff} for SAR Rev. 8 and Proposed Rev. 9

Assembly Class	Maximum k_{eff} SAR Rev. 8	Table Number in SAR Rev. 8	Maximum k_{eff} SAR Proposed Rev. 9	Table Number in Proposed Rev. 9 of the SAR
6x6A	0.7602	6.2.35	0.7888	6.2.37
6x6B	0.7611	6.2.36	0.7824	6.2.38
7x7A	0.7973	6.2.38	0.7974	6.2.40
7x7B	0.9375	6.2.19	0.9386	6.2.20
8x8A	0.7685	6.2.39	0.7697	6.2.41
8x8B	0.9368	6.2.20	0.9416	6.2.21
8x8D	0.9366	6.2.22	0.9403	6.2.23
9x9B	0.9388	6.2.25	0.9422	6.2.27
9x9D	0.9392	6.2.27	0.9394	6.2.29
9x9E	0.9406	6.2.28	0.9424	6.2.30
9x9F	0.9377	6.2.29	0.9424	6.2.31
10x10C	0.8990	6.2.32	0.9021	6.2.34
14x14C	0.9361	6.2.6	0.9400	6.2.6

<u>Containment Evaluation</u>

There is no effect of these proposed changes on the containment evaluation because the source terms used in the containment analysis are not changed.

Proposed Change No. 19

Two new fuel assembly array classes, 15x15H (PWR) and 8x8F (BWR), are added to Appendix A, Tables A.2 and A.3, respectively, as shown in Tables 19.1 and 19.2 below and in the attached marked-up CoC tables. In addition, cooling time, burnup, decay heat, and minimum initial enrichment limits are added to Table A.1, Item II.A.1.d for the 8x8F fuel assembly array/class.

Table 19.1
New Fuel Assembly Array/Class 15x15H

Fuel Assembly Array/Class	15x15H
Clad Material	Zr
Design Initial U (kg/assy.)	≤ 475
Initial Enrichment (wt % ^{235}U)	≤ 3.8
No, of Fuel Rods	208
Clad O.D. (in.)	≥ 0.414
Clad I.D. (in.)	≤ 0.3700
Pellet Dia. (in.)	≤ 0.3622
Fuel Rod Pitch (in.)	≤ 0.568
Active Fuel Length (in.)	≤ 150
No. of Guide Tubes	17
Guide Tube Thickness (in.)	≥ 0.0140

Table 19.2
New Fuel Assembly Array/Class 8x8F

Fuel Assembly Array/Class	8x8F
Clad Material	Zr
Design Initial U (kg/assy.)	$\leq$ 185
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U)	$\leq$ 4.2
Initial Maximum Rod Enrichment(wt.% ^{235}U)	$\leq$ 5.0
No. of Fuel Rods	64
Clad O.D. (in.)	$\geq$ 0.4576
Clad I.D. (in.)	$\leq$ 0.3996
Pellet Dia. (in.)	$\leq$ 0.3913
Fuel Rod Pitch (in.)	$\leq$ 0.609
Design Active Fuel Length (in.)	$\leq$ 150
No. of Water Rods	N/A
Water Rod Thickness (in.)	$\geq$ 0.0315
Channel Thickness (in.)	$\leq$ 0.055

Reason for Proposed Changes

Based on user feedback, additional fuel assemblies were identified which did not fit into any of the existing fuel assembly array/classes. Two new assembly array/classes are required to assure all user fuel types can be loaded. The 15x15H includes the B&W Mark B11 fuel design. The 8x8F represents the unique "QUAD+" assembly.

Justification for Proposed Changes

Structural Evaluation

The addition of new fuel types permitted to be transported in the HI-STAR 100 System can have an effect on the structural analyses performed in Chapter 2 if, and only if, one or more of the following occurs because of the new fuel types:

1. The design basis weights of 700 lbs (BWR) or 1680 lbs. (PWR) are exceeded.

2. The design basis temperatures are exceeded because of the presence of the new fuel types.

3. The lengths of the new fuel assemblies cause an increase in the length of the Holtec fuel spacers.

Section 2.0 of the HI-STAR SAR contains a compliance matrix showing how the structural review requirements of NUREG-1617/10 CFR Part 71 have been satisfied by the totality of analyses currently reviewed and reported in Chapter 2. To ascertain whether any of the proposed amendment items require a re-visiting of any or all of the currently approved analyses reported in Chapter 2, the Compliance Matrix was reviewed and the following conclusions reached.

1. The weights of the proposed new fuel types do not exceed the limiting (i.e., design basis) fuel weights specified in Table A.1 of Appendix A to the CoC. Therefore, no structural analysis currently approved needs to be re-visited.

2. The design basis temperatures of all components have not exceeded the values currently licensed. Therefore, no structural analyses or free thermal expansion analyses currently approved needs to be revisited.

3. The lengths of the proposed new fuel types are longer than the minimum length of the fuel assemblies currently approved for the HI-STAR 100. Therefore, the fuel spacer stability analysis in the SAR remains bounding. The lengths of the proposed new fuel types are also less than the maximum lengths specified in Table A.1 of Appendix A to the CoC.

<u>Thermal Evaluation</u>

The B&W Mark B11 and the QUAD+ fuel types have been evaluated along with the changes to existing fuel assembly array/classes as described in Proposed Change Number 16 above. These new fuel assembly array/classes are bounded by the existing design basis thermal analyses.

<u>Shielding Evaluation</u>

The accuracy of the shielding analysis is dependent upon the calculation of the radiation source term. The source term is dependent on the mass of uranium in the fuel assembly. For a specified burnup and cooling time, the radiation source term will increase as the mass of uranium increases. Minor variations in the dimensions of a fuel assembly will have a negligible impact on the radiation source term if the mass or uranium remains constant. The additional fuel assemblies proposed for the CoC are not significantly different enough from currently licensed fuel assemblies to require an assembly specific source term calculation. These new fuel assemblies are therefore bounded by the current design basis fuel assemblies. In addition, the allowable uranium mass loadings for these new fuel assemblies is specified consistent with similar fuel assemblies in the CoC thereby assuring that these assemblies are bounded by the current design basis fuel assemblies. Therefore these additions will have a negligible effect of the shielding analysis and therefore are not explicitly considered in Revision 9 of Chapter 5 of the HI-STAR SAR.

Criticality Evaluation

Criticality calculations were performed for both new fuel array/classes. The maximum k_{eff} for the 15x15H array/class was computed to be 0.9411 as shown in proposed revised SAR Table 6.2.15. The maximum k_{eff} for the 8x8F array/class was computed to be 0.9140 as shown in proposed revised SAR Table 6.2.25. Furthermore, the highest reactivity calculated for any BWR or PWR class (0.9457 for the bounding assembly in the BWR 10x10A class and 0.9478 for PWR assembly class 15x15F remains unaltered (see proposed revised SAR Tables 6.2.30 and 6.2.13, respectively). Therefore, with the proposed changes, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all authorized fuel assembly array/classes.

Containment Evaluation

The source terms used for the existing containment analysis bound those of the new fuel assembly array/classes. Therefore, there in no impact on containment.

Proposed Change No. 20

Certificate of Compliance, Appendix A.4 through A.7:

a. In Tables A.4 through A.7, "≤" signs are added to the limits in the "Decay Heat" column.

b. In Tables A.4 through A.7, the term "Note 1" is added to the table title and the following new Note 1 is added at the bottom of the table:

   Note 1: Linear interpolation between points is permitted.

c. In Table A.6, the "≥" signs for the limits in the "Assembly Burnup" column are changed to "≤" signs.

Reason for Proposed Changes

a. Decay heat limits are maximum values.

b. To allow flexibility for users to calculate intermediate limits between the limiting values shown on the tables.

c. Editorial correction. Burnup limits are maximum values.

Justification for Proposed Changes

a. and c. Editorial corrections.

b. The analyses support the assumption that these limits vary approximately linearly from point to point. Users may need the additional flexibility between cooling times to calculate more appropriate burnup, minimum enrichment, and decay heat limits.

SECTION II – PROPOSED CHANGES TO THE SAR

Proposed Change No. 21

SAR Chapter 1

a. The Holtec Design Drawings in Section 1.4 are affected by the drawing revisions proposed in storage CoC amendment request 1008-1. As described in Proposed Change Number 2 and Section III of this attachment, unique Certificate "C" drawings are being added to a new Appendix B to the CoC, which contain an appropriate level of detail for the transport CoC. Attachment 4 to this letter contains the "C" drawings. Section 1.4 of the SAR, however, will retain the latest version of the detailed drawings used for licensing and fabrication as a historical reference, similar to other SAR material. To avoid confusion, only the "C" drawings are included in this amendment request package. Please refer to storage amendment request 1008-1 for the revised detailed drawings. In addition to the Holtec drawings, the following drawings for the TN/D-1 Damaged Fuel Container and Thoria Rod Canister are added to both new CoC Appendix B and to SAR Section 1.4:

Drawing No.	Rev.	Title
9317.1-120-2	0	D-1 Canister Assembly
9317.1-120-3	1	D-1 Canister Lid Assembly
9317.1-120-4	0	D-1 Canister Body
9317.1-120-5	1	D-1 Canister Bottom Assembly
9317.1-120-6	1	D-1 Canister Lower Lid Box Assy
9317.1-120-7	0	D-1 Canister Bumper Plate
9317.1-120-8	0	D-1 Canister Bale
9317.1-120-9	1	D-1 Canister Hanger
9317.1-120-10	1	D-1 Canister Lid Box
9317.1-120-11	1	D-1 Canister Lid Frame
9317.1-120-13	0	D-1 Canister Guide Bar
9317.1-120-14	0	D-1 Canister Fuel Support Plate
9317.1-120-15	0	D-1 Canister Screen Support Plate
9317.1-120-17	0	D-1 Canister Strut
9317.1-120-18	1	D-1 Canister Screen
9317.1-120-19	2	D-1 Canister Thin (Inner) Wiper
9317.1-120-20	0	D-1 Canister Spacer
9317.1-120-21	0	D-2/3 Fuel Spacer for TN-9 Cask
9317.1-120-22	0	D-1 Canister Thick Wiper
9317.1-120-23	0	D-1 Canister Lid Assembly For Failed Fuel
9317.1-182-1	1	Thoria Rod Canister Assembly
9317.1-182-2	1	Thoria Rod Canister Spacer
9317.1-182-3	1	Thoria Rod Canister Separator Plate Assembly
9317.1-182-4	1	Thoria Rod Canister Retaining Plate Assembly
9317.1-182-5	1	Thoria Rod Canister Retaining Plate Details
9317.1-182-6	2	Thoria Rod Canister Parts List

b. Last paragraph of Section 1.0 is revised to annotate that Revision 9 is made on a page basis.

c. Table 1.0.1 in Section 1.0 is revised to 1) clarify that the classification of fuel as damaged is to be performed by the review of records, 2) add the Transnuclear Dresden Unit 1 damaged fuel canister to the definition of "damaged fuel container", 3) specify that the B_4C quantity in Holtite is nominal in lieu of minimum, and 4) correct the definition of "intact fuel" to specify that dummy fuel rods must displace a volume equal to *or greater than* the original fuel rods.

d. Subsection 1.2.1.2 is revised to clarify that material classified as Not Important to Safety may not be ASME Code material.

e. Subsections 1.2.1.2.1 and 1.2.1.2.2 are revised to add the terms "approximately" and "nominally" to the dimensions specified.

f. Subsection 1.2.1.2.2 is revised to 1) clarify the description of the aluminum heat conduction elements, 2) specify that the MPC is backfilled with a helium *pressure* in lieu of mass, 3) editorially make the number of damaged fuel container drawings in Section 1.4 plural.

g. Subsection 1.2.1.4.2 is revised to specify the quantity of B_4C as nominal in lieu of minimum.

h. Subsection 1.2.1.6 and Table 1.2.3 are revised to specify that the MPC is backfilled with a helium *pressure* in lieu of mass.

i. Subsection 1.2.3.2 is revised to correctly specify that dummy fuel rods must displace a volume equal to *or greater than* the original fuel rods.

j. Subsection 1.2.3.4 and Tables 1.2.16 and 1.2.17 are revised to clarify that the fuel spacer lengths presented are suggested and that the user must appropriately specify the correct fuel spacer lengths.

k. Subsections 1.2.3.7 and 1.2.3.8 and Tables 1.2.9, 1.2.10, 1.2.11, 1.2.13, and 1.2.21 are revised to reflect the additional fuel characteristics added by proposed change numbers 13, 14, and 16 through 19.

l. Table 1.2.19 is revised to conform with the CoC.

m. Section 1.3 is revised to clarify the requirements for PT inspection of the closure ring weld if only one weld pass is required and add the ASME Code exception that only UT or multi-layer PT is to be performed on the MPC lid to shell weld (no RT).

n. Table 1.3.3 of Section 1.3 is revised to 1) delete the backing strip, alignment pin, rupture disk coupling, and rupture disk pipe, 2) specify the material of the rupture disk as "commercial", and 3) re-classify the closure bolt washer as Not Important to Safety and stainless steel.

o. Table 1.B.1 in Appendix 1.B is revised to specify the Holtite specific gravity as nominal in lieu of maximum.

p. Appendix 1.C is revised to add NEVER-SEEZ NGBT to FEL-PRO N-5000 as acceptable materials for use on the HI-STAR 100 System.

Reason for Proposed Changes

a. The Holtec HI-STAR 100 detailed design drawings are contained in Section 1.4 and require replacement to reflect the revisions described in Attachment 4 and proposed for approval under storage amendment request 108-1. Adding the detailed drawing to the transport SAR as historical information is an administrative change. The TN/D-1 and Thoria Rod Canister drawings were used to qualify those canister designs for transportation in the HI-STAR 100 System.

b. Editorial.

c. Table 1.0.1 is revised 1) to conform with the storage certificate, 2) to add an additional damage fuel container, 3) to conform with Change 21 g. 4) editorial correction.

d. Editorial clarification to add completeness to the document.

e. Editorial clarification to add completeness to the document. The dimensional requirements are specified on the Certificate drawings of Section 1.4.

f. 1) Editorial to more accurately describe the heat conduction elements, 2) Conforming change to comply with Proposed Change Number 5, 3) Editorial.

g. The weight percentage of B_4C can vary during mixing of the material. Specifying the B_4C quantity as a nominal value provides sufficient flexibility while still meeting the design requirements.

h. Conforming change to comply with Proposed Change Number 5.

i. To correct an editorial error.

j. Due to the addition of new damaged fuel containers and minor variations in fuel assembly lengths, it is necessary to allow the user to specify the fuel spacer lengths to maintain the active fuel region adjacent to the Boral.

k. This is a conforming change with Change Numbers 13, 14, and 16 through 19.

l. To correct an editorial error and comply with the CoC.

m. This is a conforming change to comply with the CoC.

n. 1) The backing strip and alignment pins were deleted in previous revisions to the drawings and were inadvertently not deleted from Table 1.3.3 and the rupture disk coupling and rupture disk pipe have been deleted from the proposed drawings in this amendment. 2) The rupture disk is composed of multiple materials and, therefore, should be specified as commercial. 3) The correct classification for the closure bolt washer is NITS. This is an editorial correction.

o. The Holtite specific gravity will vary within a band during fabrication. Changing the term maximum to nominal will provide necessary fabrication flexibility while still meeting the design requirements.

p. Adding NEVER-SEEZ provides additional flexibility to the utilities to utilize a widely used product which has already been qualified for site use at many nuclear power plant.

Justification for Proposed Changes

a. The justification for the drawing revisions is provided in Attachment 4.

b. Editorial.

c. 1) Conforming change to comply with the revised CoC. 2) See the justification for proposed Change Number 3. 3) See the justification for proposed Change Number 21.g. 4) Conforming change to comply with the CoC.

d. Editorial clarification. Components classified as NITS do not need to be of the pedigree of ASME Code material.

e. Editorial clarification. The dimensions are specified in the Certificate drawings.

f. 1) The design and the shape of the aluminum heat conduction elements facilitate installation and conformance with the fuel basket and MPC shell. Due to the funnels facilitating the installation of the drain line, it is not possible to have a full length heat conduction element at the drain pipe location. The heat conduction element at the drain pipe location is positioned between the upper and lower funnel. This location is adjacent to the center of the fuel basket, where the maximum temperatures are predicted. The thermal analysis has not taken credit for all heat conduction elements in each basket peripheral location. 2) This is a conforming change. See the justification for proposed Change Number 4. 3) Editorial.

g. Sensitivity analyses have been performed which demonstrate that a reduction of the B_4C from 1 weight percent to 0.75 weight percent in the Holtite only causes an increase in the calculated total dose rate of approximately 3%. This increase in dose rate occurs at fuel with increased burnups due to the increase in the neutron source. For lower burnups, the effect is lessened. A 3% increase in the dose rate is close to the accuracy of the dose rate calculations. In addition, the shielding calculations assume a conservatively low density for Holtite of 1.61 gm/cc (nominal is 1.68 gm/cc) and a conservatively low hydrogen content as discussed in SAR Subsection 5.3.2. The final transport package is to be surveyed prior to shipment to demonstrate compliance with the transport radiation dose rate limits. The actual B_4C weight percent will vary in an approximate band of +/- 15% and will be confirmed by testing.

h. This is a conforming change. See the justification for Proposed Change Number 5.

i. Editorial correction to comply with the CoC.

j. The maximum fuel lengths have been structurally qualified to withstand normal conditions and the hypothetical accident conditions (60g's times the weight of the fuel assembly). Minor variations in the fuel spacer lengths must be accommodated to properly position the fuel assembly active fuel region adjacent to the Boral.

k. See proposed Change Numbers 13, 14, and 65 through 19.

l. Conforming change to comply with the CoC.

m. Editorial conforming change. This is a conforming change to comply with the CoC.

n. 1) Editorial. 2) Correction. 3) Following the guidance of NUREG-6407 and the implementing procedure at Holtec, the closure bolt washers have been classified as NITS.

o. Increasing the density increases the shielding effectiveness of the Holtite, while only providing an insignificant increase in the weight of the overpack. The density of Holtite should not exceed 1.70 g/cc, which would result in a weight increase of only approximately 200 pounds.

p. NEVER-SEEZ NGBT has been determined to be equivalent to FEL-PRO N-5000 in its lubrication properties and impurities.

<u>Proposed Change No. 22</u>

<u>SAR Chapter 2</u>

a. Table 2.0.1, under "Specific Analyses" is revised to refer to Appendices 2.B and 2.AO.

b. Table 2.2.3 is revised to correct the weight for the lift yoke to be 3,600 lb.

c. Subsection 2.10.1 is revised to add Appendix 2.AO to the list of Appendices.

d. Appendix 2.L is revised to reflect the reduction in closure ring weld size to 1/8 inch.

e. Appendix 2.O is revised to include calculations to demonstrate the adequacy of the modification to the upper fuel spacer design to accommodate certain PWR fuel assemblies (see Attachment 4, Drawing C1396, Sheet 6)

f. Appendix 2.R is revised to analyze changes to the pocket trunnion-to-overpack weld detail.

g. Appendix 2.AM has been revised to reflect the new, optional weld detail proposed for the overpack neutron shield enclosure panel to radial channel weld. The optional detail reduces the weld size from 7/16 inch to 3/16 inch.

h. New Appendix 2.AO has been added to address the TN/D-1 damaged fuel canister and Thoria Rod Canister.

Reason for Proposed Changes

a. Editorial to correct an oversight and to add reference to a new appendix.

b. Editorial to correct the value based on actual lift yoke design. The lift yoke weight is provided for information to the users to determine total load on the plant crane hook.

c. Editorial to add new Appendix 2.AO to the list of appendices.

d. Field experience with the closure ring has necessitated reducing the closure ring weld size to ensure proper fit-up and a good quality weld.

e. This is a supporting analysis for a proposed design change affecting the upper fuel spacers in the MPC.

f. The pocket trunnions-to-overpack weld was re-analyzed for a minimum weld depth of 4.125 inches.

g. The optional weld details allows users to reduce the amount of weld material and accept a surface which is irregular, rather than smooth.

h. This new appendix supports the request to include the TN/D-1 damaged fuel canister and the Thoria Rod Canister for transportation of fuel in the HI-STAR 100 System.

Justification for Proposed Changes

a. Editorial

b. Editorial. The weight of the lift yoke is not used in any cask structural analysis.

c. Editorial.

d. The smaller weld sizes provide adequate structural integrity and positive safety margins as shown in proposed revised Appendix 2.L.

e. The results of the analysis show that the shorter fuel spacer length is bounded by the longer fuel spacer length. The stresses on the new plate attachment are acceptable under all loading conditions. See Attachment 5 for proposed revised SAR Appendix 2.O for details of this analysis.

f. A minimum weld depth had previously not been specifically established for this joint. Establishing the 4.125 minimum weld depth provides clarification for the fabricator and a tie directly to the supporting structural analysis.

g. The reduction in the amount of weld material allows for a more efficient fabrication process. The 3/16 inch minimum weld meets all structural design requirements as shown in proposed revised SAR Appendix 2.AM.

h. See Section I of this attachment for justification.

<u>Proposed Change No. 23</u>

<u>SAR Chapter 3</u>

a. Section 3.3 text and Tables 3.3.2, 3.3.3, 3.3.5, 3.3.6, and 3.3.7 are revised as conforming changes in support of changes proposed in Section I of this attachment to modify/add fuel assembly array/classes. These revisions address fuel cladding stress and temperature limits for the following fuel:

B&W 15x15 Mark B-11 (Entergy-ANO)
CE-14x14 (Millstone Unit 2)
GE 6x6 Dresden-1 Fuel (with TN Damaged Fuel Container)
Dresden Unit 1 Thoria Rod Canister
GE 7x7 (GPUN-Oyster Creek)
GE 8x8 (GPUN-Oyster Creek)
GE 8x8 QUAD+ (NYPA-Fitzpatrick)
GE 8x8 (TVA-Browns Ferry)
Seimens 9x9 SPC-5 (Entergy-Grand Gulf)

b. Subsection 3.4.1.1.2 text and Tables 3.4.4 through 3.4.6 are revised, and new Table 3.4.34 is added as a conforming changes in support of changes proposed in Section I of this attachment to modify/add fuel assembly array/classes (see list in Item 'a' above).

c. New Subsection 3.4.1.1.18 is added as a conforming change in support of changes proposed in Section I of this attachment to provide a discussion of Low Heat Emitting (LHE) fuel, including the TN/D-1 damaged fuel canister and the D-1 Thoria Rod Canister.

Reason for Proposed Changes

a. This is a conforming change in support of changes to the CoC. See Section I of this attachment.

b. This is a conforming change in support of changes to the CoC. See Section I of this attachment.

c. This is a conforming change in support of changes to the CoC. See Section I of this attachment.

Justification for Proposed Changes

a. See thermal evaluation of this change in Section I of this attachment.

b. See thermal evaluation of this change in Section I of this attachment.

c. See thermal evaluation of this change in Section I of this attachment.

<u>Proposed Change No. 24</u>

<u>SAR Chapter 4</u>

a. Table 4.1.1 is revised to reduce the required pressure ratings for the overpack mechanical seals.

b. Subsections 4.2.5.7 and 4.2.5.8 are revised to correct a minor error in the MPC free gas volume used in the containment analysis.

c. Subsection 4.2.5.2 is revised to reflect the addition of thoria rods to the authorized package contents in the containment analysis.

d. Table 4.2.12 is revised to change the secondary capillary length (a) to 3.175 cm to conform with the text in Subsections 4.2.5.8 and 4.2.5.9.

Reason for Proposed Changes

a. The current design pressures stated for the overpack mechanical seals are unnecessarily high.

b. The free volumes used in the MPC-24 and MPC-68 containment analyses were incorrect.

c. This is a conforming change in support of the addition of one Thoria Rod Canister to the CoC for HI-STAR 100 transportation.

d. Editorial.

Justification for Proposed Changes

a. The design pressures currently stated in the TSAR for these seals is 10,000 psig. The maximum design internal pressure for the HI-STAR 100 System is 125 psig. Reducing the design pressures for these seals to 1,000 psig increases the selection of seal available for use while still providing a safety margin of approximately an order of magnitude.

b. This change corrects a minor error in the MPC-24 and –68 free volume used in the containment analyses. The results of the containment analyses with the incorrect (previous) free volumes yielded lower allowable leakage rates and were therefore more conservative than the revised containment analyses presented herein. There is no change to the acceptance criteria for containment boundary testing in the CoC.

c. This is a conforming change in support of the addition of a Thoria Rod Canister to the authorized contents for the HI-STAR 100 package. See Section I of this attachment for discussion of the containment analyses regarding these changes.

d. Editorial.

Proposed Change No. 25

SAR Chapter 5

a. Section 5.0 is revised to add text discussing the antimony-beryllium sources and Thoria Rod Canister which are being added as permissible contents to HI-STAR 100 for transport.

b. Section 5.1 is revised to add Dresden Unit 1 antimony-beryllium neutron sources to the list of neutron sources.

c. Subsection 5.1.1 is revised to add discussion of the Thoria Rod Canister and Dresden Unit 1 antimony-beryllium sources.

d. Subsection 5.1.2 is revised to add two words in the discussion of 10CFR71.51.

e. Subsection 5.2.1 is revised to change three of the reference numbers in Section 5.2.1. These were typographical errors.

f. Subsection 5.2.5.3 is revised to add discussion about heavy metal mass.

g. New Subsection 5.2.6 is added to discuss the source terms for the Dresden Unit 1 Thoria Rod Canister which is being added to the allowable contents for the HI-STAR 100 package.

h. New Subsection5.2.7 is added to discuss the source terms for the Dresden Unit 1 antimony-beryllium neutron sources which are being added to the allowable contents for the HI-STAR 100 package.

i. Table 5.2.1 is revised to remove the partial note from the end of the table. This was a typographical error.

j. New Tables 5.2.29 through 5.2.31 are added in support of additional Section 5.2.6.

k. Figure 5.3.9 is revised to correct a typographical error.

l. New Subsection 5.4.5 is added to discuss the source terms for the Dresden Unit 1 antimony-beryllium neutron sources which are being added to the allowable contents for the HI-STAR 100 package.

m. New Subsection 5.4.6 is added to discuss the source terms for the Dresden Unit 1 Thoria Rod Canister which is being added to the allowable contents for the HI-STAR 100 package.

n. New Table 5.4.25 is added to provide neutron source information accounting for the Sb-Be neutron source.

Reason for Proposed Changes

a. This is a conforming change to support the addition of these items for transport.

b. This is a conforming change to support the addition of this item for transport.

c. This is a conforming change to support the addition of these items for transport.

d. The new words provide a clear representation of 10CFR71.51.

e. Editorial.

f. This is a conforming change in support of increasing the limits on allowable heavy metal mass for non-design basis fuel assemblies.

g. This is a conforming change in support of permitting the transport of the Dresden Unit 1 Thoria Rod Canister.

h. This is a conforming change in support of permitting the transport of the Dresden Unit 1 antimony-beryllium neutron sources.

i. Editorial.

j. This is a conforming change in support of permitting the transport the Thoria Rod Canister.

k. Editorial.

l. This is a conforming change in support of permitting the transport of the Dresden Unit 1 antimony-beryllium neutron sources.

m. This is a conforming change in support of permitting the transport of the Dresden Unit 1 Thoria Rod Canister.

n. This is a conforming change in support of permitting the transport of the Dresden Unit 1 antimony-beryllium neutron sources.

Justification for Proposed Changes

a. See shielding evaluation in Section I of this attachment.

b. See shielding evaluation in Section I of this attachment.

c. See shielding evaluation in Section I of this attachment.

d. This is a conforming change to bring the wording in the TSAR into agreement with the regulations.

e. Correction of typographical errors.

f. See shielding evaluation in Section I of this attachment.

g. See shielding evaluation in Section I of this attachment.

h. See shielding evaluation in Section I of this attachment.

i. Correction of typographical error.

j. See shielding evaluation in Section I of this attachment.

k. Correction of typographical error.

l. See shielding evaluation in Section I of this attachment.

m. See shielding evaluation in Section I of this attachment.

n. See shielding evaluation in Section I of this attachment.

Proposed Change No. 26

SAR Chapter 6

a. Tables 6.1.1 through 6.1.3 are revised to reflect changes to the authorized fuel contents for HI-STAR 100.

b. Table numbers in section 6.2 are revised as follows:

Tables 6.2.15 through 6.2.23 become Tables 6.2.16 through 6.2.24.
Tables 6.2.24 through 6.2.39 become Tables 6.2.26 through 6.2.41.

Table references in TSAR Subsections 6.2.2, 6.2.3 and 6.2.4 are updated accordingly

c. Tables 6.2.6, 20, 21, 23, 27, 29, 30, 31, 34, 37, 38 and 41 and Appendix 6.C are revised to reflect changes to the authorized fuel contents for HI-STAR 100.

d. Tables 6.2.15 and 6.2.25 are added to Section 6.2.

e. Subsection 6.2.4, after the third sentence is revised to add "Two different DFC types with slightly different cross sections are considered." At the end of the first paragraph "for both DFC types" is added.

f. Subsection 6.2.5 is added, together with Table 6.2.42. This subsection and table provides information about the Thoria Rod Canister (see Section I of this attachment).

g. Table 6.3.4, MOX fuel specification, is revised as follows:

92235 Atom-Density from 1.659E-04 to 1.719E-04
92235 Wtg.-Fraction from 6.150E-03 to 6.380E-03
92238 Wtg.-Fraction from 8.586E-01 to 8.584E-01

h. Table 6.3.4, Specification for fuel in Thoria Rods is added.

i. Subsection 6.4.4, after the first sentence is revised to add "Two different DFC types with slightly different cross sections are considered." The third paragraph, after the first sentence is revised to add "There is no significant difference in reactivity between the two DFC types." In Table 6.4.5, a third column is added to show results of the criticality analyses for the TN/D-1 DFC.

j. Subsection 6.4.6 is added, discussing results of the criticality analyses for the Thoria Rod Canister.

k. Subsection 6.4.7 is added, discussing the impact of sealed water rods in BWR fuel assemblies on the reactivity of the cask.

l. Subsection 6.4.8 is added, discussing the impact of neutron sources in BWR fuel assemblies on the reactivity of the cask.

Reason for Proposed Changes

a. This is a conforming change to support the extended scope of fuel array classes (see Section I of this attachment).

b. This is a conforming change to support the two new fuel classes (see Section I of this attachment).

c. This is a conforming change to support the extended scope of fuel array classes (see Section I of this attachment).

e. This is a conforming change in support of adding two new fuel classes (see Section I of this attachment).

e. This is a conforming change to add the TN/D-1 Damaged Fuel Canister (see Section I of this attachment).

f. This is a conforming change to add the Thoria Rod Canister to the approved list of contents (see Section I of this attachment).

g. This is a conforming change to support the increase in the U-235 enrichment in the MOX fuel rods for fuel assembly array/class 6x6B (see Section I of this attachment).

h. This is a conforming change in support of adding the Thoria Rod Canister to the approved list of contents (see Section I of this attachment).

i. This is conforming change in support of adding the Transnuclear D-1 Damaged Fuel Canister to the authorized contents (see Section I of this attachment).

j. This is a conforming change in support of adding the Thoria Rod Canister to the approved list of contents (see Section I of this attachment).

k. This is a conforming change in support of adding BWR fuel assemblies with sealed water rods to the approved contents (see Section I of this attachment).

l. This is a conforming change in support of adding BWR assemblies with neutron sources to the approves contents (see Section I of this attachment).

Justification for Proposed Changes

These are conforming changes in support of changes to the CoC. See criticality evaluations for these changes in Section I of this attachment.

Proposed Change No. 27

SAR Chapter 7

a. Section 7.0 is revised to 1) clarify vacuum drying time information in Chapter 3, 2) clarify that DFCs must be used for damaged fuel and fuel debris as defined in the CoC, but the location where the fuel is placed in the DFC is determined by the user and 3) clarify the applicability of operational requirements for an empty overpack.

b. Subsection 7.1.1 is revised to 1) add a note clarifying the intent of the SAR operating procedures and clarifying users' responsibilities, 2) delete text referring to the need to measure the volume of water drained from the MPC for use with helium backfilling, 3) change "density" to "pressure" and 4) add clarification that root pass PT is only applicable for multi-pass welds.

c. Subsections 7.1.2, "HI-STAR 100 System Receiving and Handling Operations"; 7.1.3, "HI-STAR 100 Overpack and MPC Receipt Inspection and Loading Preparation"; 7.1.5, "MPC Closure", 7.1.6, "Preparation for Transport"; 7.2.3, "Preparation for Unloading", and 7.3 "Preparation of an Empty Package for Transport" are revised in a number of places as shown in the attached proposed SAR revisions (Attachment 5) to incorporate editorial improvements, to reflect enhancements in the operating procedures, and as conforming changes to support the CoC change of MPC helium backfill from a density limit to a pressure limit.

d. Procedural step 7.1.5.32 is revised to recognize the smaller closure ring welds.

e. Tables 7.1.1 and 7.1.2 are revised to correct a number of estimated weight values.

f. Tables 7.1.4 and 7.1.5 are revised to reflect a new description for the use of the water totalizer.

g. Table 7.1.6 is revised to correct the numbering of the inspections items.

h. Table 7.1.3 is revised to increase the torque requirement for the overpack vent and drain port plugs.

i. Subsection 7.2.3, Step 8.b is revised to reflect a different method of gaining access to the MPC vent and drain ports prior to fuel unloading.

Reason for Proposed Changes

a. Clarifications.

b. 1) Clarification. 2) and 3) These are conforming changes in support of the change from MPC helium backfill density to pressure (see Proposed Change Number 5). 4) This is a conforming change in support of the drawing change which reduces the closure ring weld size (see Attachment 4).

c. To incorporate lessons learned from dry run activities for the first production HI-STAR 100 System at Plant Hatch.

d. This is a conforming change to reflect the reduction in closure ring weld sizes shown on the drawings (see Attachment 4).

e. Editorial to correct estimated weights of certain components in the HI-STAR 100 System.

f. This is a conforming change in support of the change from MPC helium backfill density to pressure.

g. Editorial.

h. To provide sufficient compression for the seals located beneath the port plug heads.

i. This change reflects the actual equipment purchased to remove the closure ring welds.

Justification for Proposed Changes

a. 1) Clarification. Actual limits on vacuum drying time are plant-specific, based on the actual burnup and cooling time of the fuel being loaded and the temperature of the spent fuel pool water. The TSAR provides examples for illustration which assume design basis decay heat values and a range of spent fuel pool temperatures. 2) Damaged fuel and fuel debris are defined terms in the CoC. Fuel meeting either of these definitions must be placed in a DFC prior to transport in accordance with the CoC. It is the user's responsibility to control the loading of the fuel assemblies into the DFCs such that the ready-for-transport configuration meets the CoC requirements. This may be accomplished by inserting an empty DFC into the MPC and then loading the fuel assembly into the DFC. 3) Operational requirements in the SAR for empty overpacks only apply to used overpacks.

b. 1) Clarification. Procedures in the SAR are guidelines. Users are responsible for developing the detailed operating procedures. 2) and 3) These are conforming changes in support of the CoC change from MPC helium backfill density to pressure. 4) This is a conforming change in support of the drawing changes which reduce the size of the closure ring welds to 1/8 inch. Welds of this size will likely be single pass welds in which case there will be no separate root weld to perform PT examination on.

c. The proposed revisions to the operating procedures are enhancements based on actual field experience during dry run activities with the first production Hi-STAR 100 System at Plant Hatch. These changes preserve the intent of the steps they modify while providing necessary guidance and/or flexibility for implementing the step. These changes do not increase stay times in the vicinity of the cask. Therefore, the expected occupational doses for this activity will not increase. Changing from density to pressure for MPC helium backfilling is justified in Section I of this attachment (see Proposed Change Number 5).

d. This is a conforming change to reflect the smaller closure ring weld size on the design drawings.

e. Editorial. The total package weight remains less than analyzed.

f. These are conforming changes to reflect the change from MPC backfill density to pressure. The water totalizer is used in a limited role for measuring the amount of water removed from the MPC prior to helium leakage testing.

g. Editorial.

h. The seal manufacturer has recommended increasing the port plug torque to ensure sufficient compression of the seal. The depth of the seal groove machined under the heads of the port plugs ensure the seals seat at the higher torque without over-compression.

i. The weld removal system used to access the vent and drain ports during fuel unloading operations performs this operation differently, while achieving the same result. These changes do not increase stay times in the vicinity of the cask. Therefore, the expected occupational doses for this activity do not increase.

<u>Proposed Change No. 28</u>

<u>SAR Chapter 8</u>

a. Subsection 8.1.2.1, third paragraph is revised to delete the words "in accordance with ANSI N14.6" after "lifting load." Table 8.1.2 is revised to delete "per ANSI N14.6."

b. Subsection 8.1.2.2.1, third paragraph, and table 8.1.2 are revised to reflect a provision to perform the containment boundary hydrotest any time during fabrication after the containment boundary is complete, with a preference to perform it after completion of overpack fabrication, including attachment of all intermediate shells (this provision was approved during HI-STAR storage rulemaking). In addition the words "or temporary test seal" are added before "installed."

c. Subsection 8.1.2.4 is revised to change the testing for the neutron shield enclosure vessel to require a pneumatic pressure test in lieu of a bubble test. A soap bubble method is retained for use in finding any leaks discovered during the pneumatic test. See attached proposed revised SAR text in Attachment 5.

d. Subsection 8.1.3.1 and Table 8.1.2 are revised to reflect a provision to perform the Containment System Fabrication Verification Leakage test any time during fabrication after the containment boundary is complete, with a preference to perform it after completion of overpack fabrication, including attachment of all intermediate shells (this provision was approved during HI-STAR storage rulemaking). In addition, a revision is made to Subsection 8.1.3.1 to allow performance of the helium retention penetration leakage test simultaneously with the containment boundary weld leakage test.

e. Subsection 8.1.5.1 is revised regarding Holtite-A testing as shown in the attached proposed revised SAR pages to re-define the frequency of testing to be every manufactured lot instead of every mixed batch. In addition, clarify that the material composition test is required to confirm the amount of aluminum and hydrogen and that specific gravity testing is equivalent to the density confirmation test.

f. Subsection 8.1.6, first sentence is revised to replace "Each" with "The first." Conforming revisions to Table 8.1.2 are also made.

g. Table 8.1.3 is revised to delete the liquid penetrant test requirement for the root pass of the closure ring welds (3 places). In addition, a revision is made to Table 8.1.3 to allow magnetic particle testing as an option to liquid penetrant examination for the overpack intermediate shell welds (approved for HI-STAR storage during rulemaking).

Reason for Proposed Changes

a. This is a conforming change supporting Proposed Change Number 7 to delete an unnecessary reference to an ANSI Standard and make the transportation requirements for trunnion testing consistent with HI-STAR storage (CoC 1008).

b. The first change is a conforming change to make the transportation SAR consistent with the approved storage TSAR with regard to hydrotest sequencing. Allowing a suitable temporary seal for the hydrotest allows the fabricator to avoid destroying an engineered mechanical seal for a test which is not designed to verify the integrity of the seal. The hydrotest verifies the integrity of the containment boundary welds. The integrity of the closure seals is verified during helium leak testing described in SAR Section 8.1.3.1.

c. Based on experience with fabrication of the first HI-STAR production unit, fabrication efficiency is increased if a pneumatic test of the neutron shield enclosure is performed in lieu of a soap bubble test.

d. The first change is a conforming change to make the transportation SAR consistent with the approved storage TSAR with regard to helium leak testing of the containment boundary shell. Allowing the helium retention penetration leakage test to be performed simultaneously with the containment boundary leakage test improves fabrication efficiency.

e. To provide an appropriate level of testing for the Holtite material while reducing unnecessary burden on the fabricator.

f. This is a conforming change in support of a change to CoC 9261 (Proposed Change Number 8) and to make the transportation SAR consistent with the storage TSAR.

g. The first change is a conforming change based on a proposed drawing change to reduce the size of three closure ring welds to 1/8 inch. The proposed welds will not have a separate root and final pass. See Attachment 4. The second change is a conforming change to make the transportation SAR consistent with the previously approved storage TSAR.

Justification for Proposed Changes

a. SAR Subsection 8.1.2.1 provides the necessary guidance for trunnion acceptance testing and subsequent inspections. Referencing ANSI N14.6 provides unneeded additional requirements (e.g., annual testing and non-destructive examination beyond visual examination).

b. The first change was previously approved by the NRC during HI-STAR storage rulemaking and is currently allowed by the storage TSAR. This is a conforming change to make transportation consistent with storage. Allowing a suitable temporary seal for the hydrotest allows the fabricator to avoid destroying en engineered mechanical seal for a test which is not designed to verify the integrity of the seal. The hydrotest verifies the integrity of the containment boundary welds. The integrity of the closure seals is verified during helium leak testing described in SAR Section 8.1.3.1.

c. The neutron shield enclosure vessel is designed in accordance with ASME Section III, Subsection NF. Subsection NF does not require pressure testing of any kind. Pressure testing of this enclosure was a voluntary commitment. A pneumatic pressure test accomplishes the same objectives as the bubble test. A bubble solution will be used to find leaks if test pressure is unable to be held during the pneumatic test.

d. The first change was previously approved by the NRC during HI-STAR storage rulemaking and is currently allowed by the storage TSAR. This is a conforming change to make transportation consistent with storage. There are two helium tests required: one for the containment boundary shell and one for the containment boundary helium retention penetrations. Allowing the two required helium leakage tests to be performed simultaneously is acceptable because both tests are (with the previous change) now permitted to be performed at the same point during fabrication. Helium leakage at either the welds or the retention penetrations will be detected and appropriate repairs made during the single test.

e. Testing each mixed batch of Holtite is overly conservative and costly considering the controls used to mix and pour each batch. Sufficient confidence that each batch of as-poured Holtite-A is in compliance with the design requirements is provided by testing the total amount of material, (regardless of the number of mixed batches it produces) which contains the same constituent lots. Testing will be performed any time a new lot of constituent material is used in a mixed batch. Refer to the enclosed Holtec Standard Procedure HSP-107 for procedural controls imposed on Holtite-A mixing and pouring.

f. HI-STAR overpack design and fabrication are controlled by the design drawings and the fabrication procedures, including quality control inspections. The overall heat transfer capability of the overpack will be confirmed by testing of the first fabricated overpack as stated in the CoC. Additional testing of subsequent fabricated overpacks would be expected to yield similar results as the first test and is therefore deemed unnecessary.

g. The first change is a conforming change based on a drawing change proposed Attachment 4 which reduces the size of three closure ring welds to 1/8 inch. One-eighth inch welds will not have separate root and final passes. Therefore, a PT of the final pass is the appropriate inspection in addition to visual inspection. The second change is a conforming change to match HI-STAR storage. Magnetic particle examination is considered by the Code as equivalent to liquid penetrant in this application.

Section III – DRAWINGS

Proposed Change No. 29

The existing drawings referred to in CoC 9261 and contained in Section 1.4 of the HI-STAR SAR are the same drawings as those used to license HI-STAR for storage (Docket 72-1008). In the absence of an analogous process in 10 CFR Part 71 to 10 CFR 72.48, any drawing change or one time accept-as-is fabrication deviation from the current drawings renders the cask unable to be certified for dual purpose use until a Part 71 certificate amendment is approved.

Included in Attachment 4 are the revised HI-STAR 100 transportation certificate ("C") drawings. These drawings are proposed to be added to a new Appendix B to the CoC to make the CoC a stand-alone licensing document. The proposed changes to the CoC refer to this drawing set, which contain less detail, while maintaining the critical dimensions necessary to ensure the assumptions in the design analyses are preserved. The objective of the "C" drawings is to ensure dual purpose certification for all fabricated HI-STAR 100 Systems while avoiding unnecessary certificate amendment requests.

Aside from the above, the detailed drawings for storage have been proposed to be revised under storage amendment request 1008-1 (submitted concurrently with this amendment request) to correct errors and incorporate lesson learned from fabrication of the first production HI-STAR unit. Those changes that fell within the level of detail of the "C" drawings are included therein. Once the detailed design drawings are approved under storage amendment request 1008-1, a copy of those drawings will be included in transportation SAR Section 1.4 for historical purposes. This is consistent with other information in the transportation SAR which is used for licensing purposes but is not an integral part of the CoC.

For completeness, the overview of drawing changes provided with storage amendment request 1008-1 is provided here for information.

Overview of Drawing Changes

The manufacturing effort on the HI-STAR 100 prototype and the first production HI-STAR 100 (overpack and MPC-68) for Plant Hatch uncovered a number of drafting errors, discrepancies, inconsistencies, and ambiguities in the Design Drawings which had to be resolved through a laborious "Engineering Change Order" (ECO) process by Holtec and by §72.48 evaluations by Southern Nuclear (Plant Hatch). In retrospect, the need to address these changes, in large measure, was inevitable given the fact that our HI-STAR 100 submittal was an industry pioneering effort which did not have the benefit of an established precedent for the level of specificity and prescriptiveness for Design Drawings.

The revised drawings submitted with this amendment request incorporate the "lessons learned" collectively by Holtec International, Southern Nuclear, and UST&D, Inc. The revisions address fabricability issues, eliminate inconsistencies, replace unverifiable or non-essential dimensions and tolerances, incorporate operationally significant dimensions/tolerances, and remove ambiguities in the verbiage of the drawing notes. As the SFPO is aware, most of the changes noted in the drawings were submitted in the pre-certification period, but the resources required for their review were, at that time, committed to SER preparation and the decision was made by the SFPO (and accepted by Holtec) that the drawing changes for HI-STAR 100 be deferred to a post-certification amendment. In the meantime, we have been forced to implement these changes through Engineering Change Orders. Their formal incorporation in the TSAR will alleviate the need for an arduous and paper-intensive ECO and §72.48 reconciliation process *for each and every licensee who deploys Hi-STAR 100.*. While the drawing changes, indicated by numbered triangles, are self-evident, a summary of the changes is provided in the following, under the categories (i) Non-Fabricable Details, (ii) Errors or Inconsistencies, (iii) Component Fit-up, (iv) Clarifications to Eliminate Ambiguities, and (v) Design Enhancements.

i. <u>Non-Fabricable Details</u>

The following examples illustrate changes in this category:

a. Change the angular location of the four bolt holes on the bottom flange of the overpack that are used for securing the bottom shield. The pocket trunnion location interferes with the currently defined hole locations.

b. Change the overpack inner diameter to 68-9/16" (min.). The tolerances on the inner diameter can not be guaranteed in the fabricated overpack because PWHT (which follows machining of the overpack I.D.) produces dimensional changes due to strain relief, which will vary from one unit to the next.

c. Define a maximum diameter of 68-1/2" for the MPC canister and eliminate the provision for shims on the outside of the MPC. The 68-1/2" maximum dimension assures that the MPC can be installed into the overpack, whose I.D, as stated above, is prescribed as 68-9/16" (min). The elimination of external shims reduces localized weld-induced distortion which would degrade hardware quality.

d. Change radial channel-to-radial panel weld size in the overpack to 3/16" in order to eliminate heat-induced warping of the radial plates.

e. The tolerances on the overpack (minimum) cavity length and MPC external envelope length are slightly adjusted such that welding-induced dimension changes in the as-fabricated hardware would not cause a physical interference during the worst case operating scenario (the postulated transport fire event).

ii. <u>Errors or Inconsistencies</u>

Examples of the changes that fall under this category are:

a. The rupture disk joining detail (in the overpack drawing) was inconsistent with the commercially available rupture disk assemblies. The detail is revised to eliminate this inconsistency.

b. The tolerances on the O-ring grooves in the overpack were based on one manufacturer's design and may not be consistent with other manufacturers' seal designs. The seal groove geometry specification is modified to permit use of competing equivalent O-rings available in the industry.

c. For the upper fuel spacers on the MPC, the diameter of item 24 is increased from 3-3/4" to 4" to assure there is sufficient base metal to make the required weld.

iii. <u>Component Fit-Up</u>

Dimensional tolerances and tolerance stack-ups can both aid and hinder the ability to fit up two components. In many cases, fit-up is confirmed during final inspection and thus, dimensional tolerances need to be set down to ensure that as-built components will fit together. Examples of the required changes that fall under this category are:

a. Change dimensions for drain line supports to reference dimensions. Actual size will be governed by the space from the MPC canister and basket supports in the as-fabricated equipment.

b. Identify the MPC closure ring dimensions as reference values. Allow closure ring to be made from one or more pieces. Since the closure ring shall be field welded, an actual fit-up to assure high quality welds can be made only if the ring dimensions can be adjusted to best fit its location in the as-built hardware. Fit-up at the site under high radiation conditions will be more efficient with use of multiple sections rather than one monolithic ring.

c. Change hole size on 1-3/4" holes on the overpack closure plate to 1-15/16". The increased hole size will reduce the time necessary to align the closure plate holes with the top flange holes and, thus, reduce dose to the personnel engaged in the cask loading operations.

iv. <u>Clarifications to Eliminate Ambiguities</u>

Examples of the changes that fall under this category are:

a. Delete the ambiguous term "surface hardened" from items 13A and 13B on the MPC Bill of Materials and add a note to the applicable MPC drawing which states that the threads of item 13A shall be surface hardened by flash chrome plating or similar. The intent of surface hardening is to protect integral parts of a component that cannot easily be replaced. Since 13B (threaded cap) is a removable item, by hardening the integral member (13A), the surface wear on the threads will be biased towards the removable member.

b. Clarify that the stainless steel overlay thickness for the overpack inner and outer seal is a reference dimension. Actual thickness of the stainless steel overlay does not affect joint sealability. Machining after weld overlay does not allow for final verification of overlay thickness.

c. The overpack internal surface is machined after rolling and welding to assure the inner diameter is met. Localized grinding of the overpack and MPC base metal surfaces may be required to address scratches, burrs, weld spatter, and the like, which are inevitable in a manufacturing process. The allowance for local grinding is based on limiting the characteristic dimension of the ground region to the value permitted by NB-3213.10 for local membrane stresses.

v. Design Enhancements

Sophisticated fixturing, improved fabrication process (i.e., welding) and lessons learned have prompted us to incorporate a variety of enhancements into the design drawings. Examples of the changes that fall under this category are:

a. An alternative design for MPC basket supports allows the fabricator improved dimensional control over the clearance between the basket supports welded to the inside surface of the MPC canister and the fuel basket over the length of the canister.

b. Eliminate MPC basket shims. Shims were permitted to give added flexibility to the manufacturer. Manufacturing experience gathered to date shows that the fixturing assures dimensions can be met and, thus, shimming is not required. Removal of shims will eliminate localized heat distortion.

c. Allow option to change sheathing weld (a non-structural weld) length and pitch from two-inch long welds at eight-inch pitches to one-inch long welds at four-inch pitches. The total amount of weld remains the same; however, the revised spacing minimizes the extent of waviness in the sheathing caused by basket panel welding.

d. In the overpack, the pocket trunnions are welded using a qualified weld procedure, directly to the overpack bottom plate, eliminating the requirement for overlay on the pocket trunnion. Elimination of the overlay reduces the distortion of the machined semi-circular recess in the trunnion, which serves as the bearing surface during the rotation operation on the overpack. The trunnion made of precipitation hardened stainless steel will receive the Code-specified hardening heat treatment at 1150° when the vessel is post-weld heat treated.

The above list, of course, is not exhaustive. It does, however, help illustrate the necessity and advisability of incorporating the changes to the Design Drawings through the amendment request submitted herewith.

ATTACHMENT 2

MARK-UPS OF
PROPOSED CHANGES TO COC 9261
INCLUDING APPENDIX A
(STRIKEOUT/ITALIC FORMAT)

'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	*1* ~~0~~	USA/9261/B(U)F-85	1	7

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*)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION:

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 which house the spent nuclear fuel and an overpack which 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 203 1/8 inches without impact limiters and approximately 305 7/8 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is approximately ~~280,000~~ *282,000* pounds. Specific tolerances are called out in drawings listed below.

Multi-Purpose Canister

There are three Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions. A single overpack design is provided which is capable of storing 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 is designed to contain up to 24 Pressurized Water Reactor (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.

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 MPC is fixed. However, the number of spent fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. For the HI-STAR 100 System transporting fuel debris in a MPC-68F, 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 drawings listed below which are found in ~~Section 1.4 of Revision 8 of the Holtec HI-STAR Safety Analysis Report (SAR), Rev. 8)~~ *Appendix B to this Certificate of Compliance.*

(a) Drawing C1395, Sheets 1- *4*, Revision *1* ~~10~~
~~Sheet 2-3, Revision 9~~
~~Sheet 4, Revision 8~~

(b) Drawing C1396, Sheets 1-*4*, *6*, Revision *1* ~~12~~
Sheet *5* ~~s 2-3,~~ Revision ~~9~~ *0*
~~Sheets 4-5, Revision 8~~
~~Sheet 6, Revision 7~~

(c) Drawing C1397, Sheets 1-*4*, *6*, *7*, Revision *1* ~~14~~
Sheet *5* ~~s 2-3,~~ Revision ~~9~~ *0*
~~Sheets 4, Revision 11~~
~~Sheets 5-7, Revision 8~~

(d) Drawing C1398, Sheets 1-*3*, Revision *1* ~~12~~
~~Sheet 2, Revision 9~~
~~Sheet 3, Revision 8~~

(e) Drawing C1399, Sheets 1-*3*, Revision *1* ~~10~~
~~Sheet 2, Revision 8~~
~~Sheet 3, Revision 9~~

(f) Drawing C1401, Sheets 1-*4*, Revision *1* ~~11~~
~~Sheets 2 and 4, Revision 8~~
~~Sheet 3, Revision 9~~

(g) Drawing C1402, Sheets 1-*4*, *6*, Revision *1* ~~13~~
Sheet *5* ~~s 2-3,~~ Revision ~~9~~ *0*
~~Sheets 4-5, Revision 9~~
~~Sheet 6, Revision 7~~

(h) Drawing C1765, Sheets 1-*6*, Revision *1* ~~12~~
Sheet *7* ~~s 2-3,~~ Revision ~~9~~ *0*
~~Sheet 3, Revision 5~~
~~Sheet 4, Revision 10~~
~~Sheets 5 and 7, Revision 4~~
~~Sheet 6, Revision 1~~

(i) Drawing C1782, Revision 1

(j) Drawing C1783, Revision 1

(k) Drawing C1784, Revision ~~0~~ *1*

(l) Drawing BM-C1476, Sheets *1 & 2*, Revision *1* ~~12~~
~~Sheet 2, Revision 13~~

(m) Drawing BM-C1478, Sheets 1& 2, Revision *1* ~~9~~
~~Sheet 2, Revision 13~~

(n) Drawing BM-C1479, Sheets 1& 2, Revision *1* ~~10~~
~~Sheet 2, Revision 13~~

(o) Drawing BM-C1819, Revision 1

(p) *Drawings 9317.1-120-2, 4, 7, 8, 13-17, and 20-23, Revision 0*

(q) *Drawings 9317.1-120-3, 5, 6, 9, 10, 11, and 18, Revision 1*

***Drawings** (continued)*

***(r)** Drawing 9317.1-120-19, Revision 2*

***(s)** Drawings 9317.1-182-1 through 5, Revision 1*

***(t)** Drawings 9317.1-182-6, Revision 2*

5.b. Contents of Packaging

(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 items 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, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. ~~A damaged fuel assembly's inability to~~ *Fuel assemblies which cannot* be handled by normal means ~~must be~~ due to ~~mechanical damage and must not be due to~~ fuel ~~rod~~ cladding damage *are considered fuel debris.*

Damaged Fuel Containers are specially designed fuel containers for damaged fuel assemblies or fuel debris which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. *DFCs authorized for use in the HI-STAR 100 System are the Holtec design or the Transnuclear Dresden Unit 1 design as shown on the applicable design drawings in the HI-STAR 100 Safety Analysis Report.*

Fuel Debris ~~refers to~~ *is* ruptured fuel rods, severed rods, ~~and~~ loose fuel pellets *or fuel assemblies* with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

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. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, 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.

Planar-Average Initial Enrichment is the ~~simple~~ average of the distributed fuel rod *initial* enrichments within a given axial plane of the assembly lattice.

(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 Zircaloy clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining Zircaloy 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, 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, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR control rods, burnable poison rod assemblies, thimble plugs, and other non-fuel hardware are not authorized for transportation.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

5.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 writ operating procedures. Procedures for both preparation and operation shall be developed. ~~at~~ *At* a minimum, those procedures shall include the following provisions:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

(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 leak rate of not greater than 4.3×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} std cm^3/sec (helium) and shall be performed:

(i) before the first shipment;
(ii) within the 12-month period prior to each successive shipment;
(iii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iv) After each seal replacement.

(b) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} std 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 containment boundary seal must be replaced after each use of the seal.

(4) The rupture discs on the neutron shield vessel shall be replaced every 5 years.

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than 5×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} std cm^3/sec (helium).

(6) Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

(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.

(7) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium: ~~0.1212 g-moles/l (+0,-10%)~~ *≤ 28.3 psig* for the MPC-24 and ~~0.1218 g-moles/l (+0,-10%)~~ *≤ 28.5 psig* for the MPC-68 and MPC-68F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack cavity 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	2895 ± 90~~**~~
Overpack Vent and Drain Port Plugs	~~22 +2/-0~~ *45 +5/-0*
Top Impact Limiter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0
Tie-down Bolts	250 +20/-0
Transport Frame Bolts	250 +20/-0

~~**Tighten closure plate bolts in 5 passes and in a crisscross pattern.~~

(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. ~~in accordance with ANSI N14.6~~.

(2) The MPC shall be pressure tested to 125% of the design pressure. The minimum test pressure shall be 125 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 process, including findings (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 performed at a minimum of 12 locations on the radial neutron shield and at a minimum of 4 locations on each impact limiter

(7) ~~Each~~ *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. ~~8~~ *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.

(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 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-68, and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be *verified by* chemistry or neutron attenuation techniques.

(10) The minimum flux trap size for the MPC-24 is 1.09 inches

(11) The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

(12) The package containment verification leak test shall be per ANSI 14.5.

7. The maximum gross weight of the package as presented for shipment shall not exceed ~~280,000~~ *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* 6 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

Attachment: Appendix A

REFERENCES:

The drawings specified in this certificate 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 ~~8~~ *9*.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 31, 1999

Table A.1
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:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 1.1-2 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, *decay heat* and minimum *initial* enrichment per assembly	
i. Zr Clad:	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.
d. Fuel assembly length:	≤ 176.8 inches (nominal design)
e. Fuel assembly width:	≤ 8.54 inches (nominal design)
f. Fuel Assembly Weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain control components.

D. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

Table A.1 (continued)
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:	Zircaloy (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, *decay heat* and minimum *initial* enrichment per assembly:	
i. Zr Clad:	An assembly post-irradiation cooling time, average burnup, *decay heat* and minimum *initial* enrichment as specified in Table A.7, except for array/class 6x6A, 6x6C, and 8x8A fuel assemblies, which shall have a cooling $\geq$ 18years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U *and array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, a decay heat $\leq$ 183.5 Watts, 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. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)

Table A.1 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	*Zircaloy (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* ~~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 with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68.

Table A.1 (continued)
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:	Zircaloy (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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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 after discharge $\geq$ 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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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.
f. 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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

7. *Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:*

a. Cladding Type:	*Zircaloy (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 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; ~~or~~
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 with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68.

Table A.2
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	15x15A
Clad Material (Note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.) (Note *3*)	~~≤ 402~~ *≤ 407*	~~≤ 402~~ *≤ 407*	~~≤ 410~~ *≤ 425*	≤ 400	~~≤ 420~~ *≤ 464*
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	~~≤ 0.3840~~ *≤ 0.3880*	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	~~≤ 0.3770~~ *≤ 0.3805*	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5 (Note ~~3~~ *4*)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	~~≥ 0.040~~ *≥ 0.038*	≥ 0.0145	≥ 0.0165

~~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. Each guide tube replaces 4 fuel rods.~~

~~D. Design initial uranium weight is the nominal weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total initial uranium weight may be up to 2.0 percent higher than the design initial uranium weight due to manufacturer tolerances.~~

Table A.2 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (wt % ^{235}U)	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1
No. of Fuel Rods	204	204	208	208	208
Clad O.D. (in.)	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Clad I.D. (in.)	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Pellet Dia. (in.)	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	21	17	17	17
Guide Tube Thickness (in.)	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

~~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 weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total initial uranium weight may be up to 2.0 percent higher than the design initial uranium weight due to manufacturer tolerances.~~

Table A.2 (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 ~~4~~ 3)	≤ 420	*≤ 475*	~~≤ 430~~ *≤ 443*	~~≤ 450~~ *≤ 467*	~~≤ 464~~ *≤ 467*	~~≤ 460~~ *≤ 474*
Initial Enrichment (wt % ^{235}U)	≤ 4.0	*≤ 3.8*	≤ 4.6	≤ 4.0	≤ 4.0	≤ 4.0
No. of Fuel Rods	204	*208*	236	264	264	264
Clad O.D. (in.)	≥ 0.422	*≥ 0.414*	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Clad I.D. (in.)	≤ 0.3890	*≤ 0.3700*	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
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 Tubes	21	*17*	5 (Note ~~3~~ *4*)	25	25	25
Guide Tube Thickness (in.)	≥ 0.0145	*≥ 0.0140*	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values 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 ~~initial~~ uranium weight *limit specified in this table* may be *increased* up to 2.0 percent ~~higher than the design initial uranium weight due~~ *for comparison with users' fuel records* to *account for* manufacturer tolerances.
4. *Each guide tube replaces four fuel rods.*

Table A.3
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 ~~4~~ *3*)	~~≤ 108~~ *≤ 110*	~~≤ 108~~ *≤ 110*	~~≤ 108~~ *≤ 110*	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note ~~3~~ *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	~~≤ 4.0~~ *≤ 5.5*	≤ 5.0	≤ 4.0
No. of Fuel Rods	*35 or* 36	*35 or* 36 (up to 9 MOX rods)	36	49	49	*63 or* 64
Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Clad I.D. (in.)	~~≤ 0.4945~~ *≤ 0.5105*	≤ 0.4945	≤ 0.4990	~~≤ 0.4200~~ *≤ 0.4204*	≤ 0.4990	≤ 0.3620
Pellet Dia. (in.)	~~≤ 0.4940~~ *≤ 0.4980*	≤ 0.4820	≤ 0.4880	≤ 0.4110	~~≤ 0.4880~~ *≤ 0.4910*	≤ 0.3580
Fuel Rod Pitch (in.)	~~0.694~~ *≤ 0.710*	~~0.694~~ *≤ 0.710*	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	~~≤ 110~~ *≤ 120*	~~≤ 110~~ *≤ 120*	≤ 77.5	~~≤ 79~~ *≤ 80*	≤ 150	~~≤ 110~~ *≤ 120*
No. of Water Rods *(Note 11)*	*1 or* 0	*1 or* 0	0	0	0	*1 or* 0
Water Rod Thickness (in.)	~~N/A~~ *> 0*	~~N/A~~ *> 0*	N/A	N/A	N/A	~~N/A~~ *≥ 0*
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

~~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. ≤ 0.612 wt. % ^{235}U and ≤ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu).~~

~~4. 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 initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.~~

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	*8x8F*	9x9A	9x9B
Clad Material (Note 2)	Zr	Zr	Zr	Zr	*Zr*	Zr	Zr
Design Initial U (kg/assy.) (Note ~~6~~ 3)	≤ 185	≤ 185	≤ 185	~~≤ 180~~ *≤ 185*	*≤ 185*	~~≤ 173~~ *≤ 177*	~~≤ 173~~ *≤ 177*
Maximum planar average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	*≤ 4.2*	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	*≤ 5.0*	≤ 5.0	≤ 5.0
No. of Fuel Rods	63 *or 64*	62	60 *or 61*	59	*64*	74/66 (Note ~~3~~ 5)	72
Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	*≥ 0.4576*	≥ 0.4400	≥ 0.4330
Clad I.D. (in.)	~~≤ 0.4250~~ *≤ 0.4295*	≤ 0.4250	~~≤ 0.4190~~ *≤ 0.4230*	≤ 0.4250	*≤ 0.3996*	≤ 0.3840	≤ 0.3810
Pellet Dia. (in.)	~~≤ 0.4160~~ *≤ 0.4195*	≤ 0.4160	~~≤ 0.4110~~ *≤ 0.4140*	≤ 0.4160	*≤ 0.3913*	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	~~0.636-0.641~~ *≤ 0.642*	~~0.636-0.641~~ *≤ 0.641*	≤ 0.640	≤ 0.640	*≤ 0.609*	≤ 0.566	~~0.569~~ *≤ 0.572*
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	*≤ 150*	≤ 150	≤ 150
No. of Water Rods *(Note 11)*	1 *or 0*	2	1 - 4 (Note ~~5~~ 7)	5	*N/A (Note 12)*	2	1 (Note ~~4~~ *6*)
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	*≥ 0.0315*	> 0.00	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	*≤ 0.055*	≤ 0.120	≤ 0.120

~~Notes: 1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.~~

~~4. Square, replacing nine fuel rods.~~

~~5. Variable~~

~~6. 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 initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.~~

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9C	9x9D	9x9E *(Note 13)*	9x9F *(Note 13)*	10x10A
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note ~~4~~ *3*)	~~≤ 173~~ ≤ *177*	~~≤ 170~~ ≤ *177*	~~≤ 170~~ ≤ *177*	~~≤ 170~~ ≤ *177*	~~≤ 182~~ ≤ *186*
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	~~≤ 4.2~~ ≤ *4.1*	~~≤ 4.2~~ ≤ *4.1*	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note ~~3~~ *8*)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	~~≤ 0.3590~~ ≤ *0.3640*	~~≤ 0.3810~~ ≤ *0.3860*	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	~~≤ 0.3525~~ ≤ *0.3530*	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods *(Note 11)*	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	~~> 0.0305~~ ≥ *0.0300*	~~> 0.0305~~ ≥ *0.0120*	~~> 0.0305~~ ≥ *0.0120*	≥ 0.0300
Channel Thickness (in.)	≤ 0.100	≤ 0.100	~~≤ 0.100~~ ≤ *0.120*	~~≤ 0.100~~ ≤ *0.120*	≤ 0.120

~~Notes: 1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 92 total fuel rods, 78 full length rods and 14 partial length rods.~~

~~4. 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 initial uranium weight may be up to 1.5 percent higher than the design initial uranium weight due to manufacturer tolerances.~~

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note ~~6~~ *3*)	~~≤ 182~~ *≤ 186*	~~≤ 180~~ *≤ 186*	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5
No. of Fuel Rods	91/83 (Note ~~3~~ *9*)	96	100	96
Clad O.D. (in.)	≥ 0.3957	~~≥ 0.3790~~ *≥ 0.3780*	≥ 0.3960	≥ 0.3940
Clad I.D. (in.)	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Pellet Dia. (in.)	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods *(Note 11)*	1 (Note ~~4~~ *6*)	5 (Note ~~5~~ *10*)	0	4
Water Rod Thickness (in.)	> 0.00	~~≥ 0.034~~ *≥ 0.031*	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Notes:
1. ~~Initial uranium weights and all~~ *All* dimensions are design nominal values. ~~Actual uranium weights may be higher, within the manufacturer's tolerance.~~ 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. ≤ ~~0.612~~ *0.635* wt. % ^{235}U and ≤ 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 75 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 non-fissile 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 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS *(Note 1)*

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 10	≤ 24,500	≥ 2.3	≤ 411
≥ 12	≤ 29,500	≥ 2.6	≤ 473
≥ 14	≤ 34,500	≥ 2.9	≤ 540
≥ 15	≤ 37,500	≥ 3.2	≤ 579

Note 1: Linear interpolation between points is permitted.

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS *(Note 1)*

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 7	≤ 24,500	≥ 2.3	≤ 496
≥ 8	≤ 29,500	≥ 2.6	≤ 562
≥ 10	≤ 34,500	≥ 2.9	≤ 610
≥ 12	≤ 39,500	≥ 3.2	≤ 667
≥ 15	≤ 44,100	≥ 3.4	≤ 704

Note 1: Linear interpolation between points is permitted.

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH STAINLESS STEEL CLAD *(Note 1)*

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 19	~~≥~~ ≤ 30,000	≥ 3.1	≤ 377
≥ 24	~~≥~~ ≤ 40,000	≥ 3.1	≤ 475

Note 1: Linear interpolation between points is permitted.

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68 *(Note 1)*

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 8	≤ 24,500	≥ 2.1	≤ 179
≥ 9	≤ 29,500	≥ 2.4	≤ 208
≥ 12	≤ 34,500	≥ 2.6	≤ 222
≥ 15	≤ 39,100	≥ 2.9	≤ 238

Note 1: Linear interpolation between points is permitted.

ATTACHMENT 3

PROPOSED REVISED COC 9261 INCLUDING APPENDIX A (FINAL FORM)

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	1	USA/9261/B(U)F-85	1	7

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 which house the spent nuclear fuel and an overpack which 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 203 1/8 inches without impact limiters and approximately 305 7/8 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is approximately 282,000 pounds. Specific tolerances are called out in drawings listed below.

Multi-Purpose Canister

There are three Multi-Purpose Canister (MPC) models, designated the MPC-24, MPC-68, and MPC-68F. All MPCs are designed to have identical exterior dimensions. A single overpack design is provided which is capable of storing 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 is designed to contain up to 24 Pressurized Water Reactor (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.

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 MPC is fixed. However, the number of spent fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. For the HI-STAR 100 System transporting fuel debris in a MPC-68F, 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 drawings listed below which are found in Appendix B to this Certificate of Compliance.

(a) Drawing C1395, Sheets 1-4, Revision 1

(b) Drawing C1396, Sheets 1-4, 6, Revision 1
Sheet 5, Revision 0

(c) Drawing C1397, Sheets 1-4, 6, 7, Revision 1
Sheet 5, Revision 0

(d) Drawing C1398, Sheets 1-3, Revision 1

(e) Drawing C1399, Sheets 1-3, Revision 1

(f) Drawing C1401, Sheets 1-4, Revision 1

(g) Drawing C1402, Sheets 1-4, 6, Revision 1
Sheet 5, Revision 0

(h) Drawing C1765, Sheets 1-6, Revision 1
Sheet 7, Revision 0

(i) Drawing C1782, Revision 1

(j) Drawing C1783, Revision 1

(k) Drawing C1784, Revision 1

(l) Drawing BM-C1476, Sheets 1 & 2, Revision 1

(m) Drawing BM-C1478, Sheets 1& 2, Revision 1

(n) Drawing BM-C1479, Sheets 1& 2, Revision 1

(o) Drawing BM-C1819, Revision 1

(p) Drawings 9317.1-120-2, 4, 7, 8, 13-17, and 20 -23, Revision 0

(q) Drawings 9317.1-120-3, 5, 6, 9, 10, 11, and 18 Revision 1

(r) Drawing 9317.1-120-19, Revision 2

(s) Drawings 9317.1-182-1 through 5, Revision 1

(t) Drawing 9317.1-182-6, Revision 2

5.b. Contents of Packaging

(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 items 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, missing fuel rods that are not replaced 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.

Damaged Fuel Containers are specially designed fuel containers for damaged fuel assemblies or fuel debris which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STAR 100 System are the Holtec design or the Transnuclear Dresden Uit 1 design as shown on the applicable design drawings in the HI-STAR 100 Safety Analysis Report.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

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. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, 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.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

(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 Zircaloy clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining Zircaloy 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, 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, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies.

(f) PWR control rods, burnable poison rod assemblies, thimble plugs, and other non-fuel hardware are not authorized for transportation.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

5.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:

(1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

(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 leak rate of not greater than 4.3×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.15×10^{-6} std cm^3/sec (helium) and shall be performed:

(i) before the first shipment;
(ii) within the 12-month period prior to each successive shipment;
(iii) after detensioning one or more overpack lid bolts or the vent port plug; and
(iv) After each seal replacement.

(b) Before each shipment, all containment boundary seals shall be leak tested using a test with a minimum sensitivity of 1×10^{-3} std 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 containment boundary seal must be replaced after each use of the seal.

(4) The rupture discs on the neutron shield vessel shall be replaced every 5 years.

(5) All MPCs shall be leak tested at the time of closure to show a leak rate of no greater than 5×10^{-6} std cm^3/sec (helium). The leak test shall have a minimum sensitivity of 2.5×10^{-6} cm^3/sec (helium).

(6) Water and residual moisture shall be removed from the MPC in accordance with the following specifications:

(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.

(7) Following vacuum-drying, the MPC shall be backfilled with 99.995% minimum purity helium: $\leq$ 28.3 psig for the MPC-24 and $\leq$ 28.5 psig for the MPC-68 and MPC-68F.

(8) Water and residual moisture shall be removed from the HI-STAR 100 overpack in accordance with the following specifications:

(a) The MPC shall be evacuated to a pressure of less than or equal to 3 torr.

(b) The overpack cavity 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	2895 $\pm$ 90
Overpack Vent and Drain Port Plugs	45 +5/-0
Top Impact Lim iter Attachment Bolts	256 +10/-0
Bottom Impact Limiter Attachment Bolts	1500 +45/-0
Tie-down Bolts	250 +20/-0
Transport Frame Bolts	250 +20/-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.

(2) The MPC shall be pressure tested to 125% of the design pressure. The minimum test pressure shall be 125 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 process, including findings (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 performed at a minimum of locations on the radial neutron shield and at a minimum of 4 locations on each impact limit

(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.

(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 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-68, and 0.01 g/cm^2 for the MPC-68F. The ^{10}B loading shall be verified by chemistry or neutron attenuation techniques.

(10) The minimum flux trap size for the MPC-24 is 1.09 inches

(11) The minimum fuel cell pitch for the MPC-68 and MPC-68F is 6.43 inches.

(12) The package containment verification leak test shall be per ANSI 14.5.

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 6 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

Attachment: Appendix A

REFERENCES:

The drawings specified in this certificate 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 9.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: March 31, 1999

Table A.1
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:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 1.1-2 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, decay heat and minimum initial enrichment per assembly	
i. Zr Clad:	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.
d. Fuel assembly length:	$\leq$ 176.8 inches (nominal design)
e. Fuel assembly width:	$\leq$ 8.54 inches (nominal design)
f. Fuel Assembly Weight:	$\leq$ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain control components.

D. Damaged fuel assemblies and fuel debris are not authorized for loading into the MPC-24.

Table A.1 (continued)
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:	Zircaloy (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, decay heat and minimum initial enrichment per assembly:	
i. Zr Clad:	An assembly post-irradiation cooling time, average burnup, decay heat and minimum initial enrichment as specified in Table A.7, except for array/class 6x6A, 6x6C, 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 array/class 8x8F fuel assemblies, which shall have a cooling time $\geq$ 10 years, an average burnup $\leq$ 27,500 MWD/MTU, a decay heat $\leq$ 183.5 Watts, 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. Fuel assembly length:	$\leq$ 176.2 inches (nominal design)
f. Fuel assembly width:	$\leq$ 5.85 inches (nominal design)

g. Fuel assembly weight	≤ 700 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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	≤ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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 (continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (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:	Zircaloy (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	≤ 400 lbs, including channels

Table A.1(continued)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	Zircaloy (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 Assembly:	A fuelpost-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 with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68.

Table A.1 (continued)
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:	Zircaloy (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 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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 after discharge ≥ 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	≤ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1 (continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (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:	Zircaloy (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	$\leq$ 400 lbs, including channels

Table A.1(continued)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	Zircaloy (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 Assembly:	An assembly 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 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 with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68.

Table A.2
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	14x14A	14x14B	14x14C	14x14D	15x15A
Clad Material (Note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 464
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5 (Note 4)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	≥ 0.0165

Table A.2 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	15x15B	15x15C	15x15D	15x15E	15x15F
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (wt % ^{235}U)	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1
No. of Fuel Rods	204	204	208	208	208
Clad O.D. (in.)	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Clad I.D. (in.)	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Pellet Dia. (in.)	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide Tubes	21	21	17	17	17
Guide Tube Thickness (in.)	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (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 (wt % ^{235}U)	≤ 4.0	≤ 3.8	≤ 4.6	≤ 4.0	≤ 4.0	≤ 4.0
No. of Fuel Rods	204	208	236	264	264	264
Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
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 Tubes	21	17	5 (Note 4)	25	25	25
Guide Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Notes:

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

B. Zr. Designates cladding material made of Zirconium or Zirconium alloys.

C. Design initial uranium weight is the 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.

D. Each guide tube replaces four fuel rods.

Table A.3
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 Rods	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
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 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A	9x9B
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	63 or 64	62	60 or 61	59	64	74/66 (Note 5)	72
Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400	≥ 0.4330
Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840	≤ 0.3810
Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 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	1 (Note 6)
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120	≤ 0.120

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	10x10A
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 186
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.1	≤ 4.1	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note 8)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0300
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	10x10B	10x10C	10x10D	10x10E
Clad Material (Note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5
No. of Fuel Rods	91/83 (Note 9)	96	100	96
Clad O.D. (in.)	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Clad I.D. (in.)	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Pellet Dia. (in.)	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

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. ≤ 0.635 wt. % ^{235}U and ≤ 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 75 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 non-fissile 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 PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS (Note 1)

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 10	≤ 24,500	≥ 2.3	≤ 411
≥ 12	≤ 29,500	≥ 2.6	≤ 473
≥ 14	≤ 34,500	≥ 2.9	≤ 540
≥ 15	≤ 37,500	≥ 3.2	≤ 579

Note 1: Linear interpolation between points is permitted.

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS (Note 1)

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 7	≤ 24,500	≥ 2.3	≤ 496
≥ 8	≤ 29,500	≥ 2.6	≤ 562
≥ 10	≤ 34,500	≥ 2.9	≤ 610
≥ 12	≤ 39,500	≥ 3.2	≤ 667
≥ 15	≤ 44,100	≥ 3.4	≤ 704

Note 1: Linear interpolation between points is permitted.

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-24 PWR FUEL WITH STAINLESS STEEL CLAD (Note 1)

Post-Irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 19	≤ 30,000	≥ 3.1	≤ 377
≥ 24	≤ 40,000	≥ 3.1	≤ 475

Note 1: Linear interpolation between points is permitted.

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT
MPC-68 (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Minimum Enrichment (wt. % U-235)	Decay Heat (Watts)
≥ 8	≤ 24,500	≥ 2.1	≤ 179
≥ 9	≤ 29,500	≥ 2.4	≤ 208
≥ 12	≤ 34,500	≥ 2.6	≤ 222
≥ 15	≤ 39,100	≥ 2.9	≤ 238

Note 1: Linear interpolation between points is permitted.

BILL OF MATERIALS FOR HI-STAR 100 OVERPACK (BM-C1476)

REF. DWGS. C1397,C1398 & C1399.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-18-99 INCORPORATED ECO/ SMDR/MISC CHANGES	11/24/99	B.G. 11/24/99	11/24/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1	1	SA-350 LF3	12" THK. BASE PLATE	BOTTOM PLATE
2	1	SA-203-E	2 1/2" THK. PLATE	INNER SHELL
3	20	SA-515 GRADE 70	1/2" THK. PLATE	ENCLOSURE SHELL PANELS
4	20	SA-515 GRADE 70	1/2" THK.	RADIAL CHANNELS
5	2	SA-705 630 17-4 PH OR SA-564 630 17-4 PH	FORGING	POCKET TRUNNION
6	4	SA-193 GRADE B7	SOCKET SET SCREW	CLOSURE PLATE PLUG
7	2	SB-637-N07718	BAR	LIFTING TRUNNION
8	1	SA-350 LF3	FORGING	TOP FLANGE
9	2	SA-203-E OR SA-350-LF3	1 1/2" THK.	REMOVEABLE SHEAR RING
10	1	SA-350 LF3	6" THK.	CLOSURE PLATE
11	2	SB-637-N07718	1 5/8" - 8 UN CAP SCREW	CLOSURE PLATE SHORT BOLT
12	1	SA-516 GRADE 70	1 1/4" THK. PLATE	INTERMEDIATE SHELL #1
13	1	SA-516 GRADE 70	1 1/4" THK. PLATE	INTERMEDIATE SHELL #2
14	1	SA-516 GRADE 70	1 1/4" THK. PLATE	INTERMEDIATE SHELL #3
15	1	SA-516 GRADE 70	1 1/4" THK. PLATE	INTERMEDIATE SHELL #4
16	1	SA-516 GRADE 70	1" THK. PLATE	INTERMEDIATE SHELL #5
17	2	SA-515 GRADE 70	1/2" THK. PLATE	ENCLOSURE SHELL RETURN
18	2	SA-193 GRADE B8	BAR	PORT PLUG
19	3	ALLOY X750	SPRING ENERGIZED SEAL,	PORT PLUG SEAL
20	4	SA-193 GRADE B7	SOCKET CAP SCREW	TRUNNION LOCKING PAD BOLT
21	2	SA-516 GRADE 70	1/2" THK. PLATE	LIFTING TRUNNION END CAP
22	4	SA-193 GRADE B7	BOLTS	TRUNNION END CAP BOLT
23	2	SA-516 GRADE 70	3/8" THK. PLATE	LIFTING TRUNNION LOCKING PAD
24	AS REQ.	HOLTITE - A	HOLTITE-A	NEUTRON SHIELD
25	8	SA-193 GRADE B7	SOCKET SET SCREW	REMOVEABLE SHEAR RING PLUG
26	1	COMMERCIAL	SELF ENERGIZED SEAL	CLOSURE PLATE INNER SEAL

BILL OF MATERIALS FOR HI-STAR 100 OVERPACK (BM-C1476)

REF. DWGS. C1397,C1398 & C1399.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-2-99 INCORPORATED ECO/ SMOR/MISC CHANGES	11/24/99	B.G 11/24/99	MS 11/24/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
27	1	COMMERCIAL	SELF ENERGIZED SEAL.	CLOSURE PLATE OUTER SEAL
28	2	SA-350-LF3 OR SA-203-E	1 1/2" THK. PLATE	PORT COVER
29	8	SA-193 GRADE B7	3/8 - 16 UNC SCREW	PORT COVER BOLT
30	2	ALLOY X750	SPRING ENERGIZED SEAL.	PORT COVER SEAL
31	--	---	DELETED	---
32	52	SB-637-N07718	1 5/8" - 8 UN CAP SCREW	CLOSURE PLATE LONG BOLT
33	2 (MIN.)	COMMERCIAL	RUPTURE DISK	RUPTURE DISK
34	8	SA-193 GRADE B7	3/8" - 16 UNC SCREW	REMOVEABLE SHEAR RING BOLT
35	1	SA-193 GRADE B8	7/8" Ø BAR	DRAIN PORT PLUG
36	--	----	DELETED	----
37	AS REQ.	SILICONE FOAM	TYPE HT-870 (BISCO PRODUCTS) OR EQUIVALENT	THERMAL EXPANSION FOAM
38	--	---	DELETED	---
39	2	SA-516 GRADE 70 OR A569	11 GAGE (1/8" THK.)	RUPTURE DISK PLATE
40	1	SA 240 304	14 GAGE (0.0751" THK.) SHEET	STORAGE MARKING NAME PLATE
41	1	SA 240 304	14 GAGE (0.0751" THK.) SHEET	TRANSPORTATION MARKING NAME PLATE
42	AS REQD	SA515-70	AS REQUIRED	BRIDGE
43	2	SA 240 304	11 GAGE (1/8" THK.) SHEET	POCKET TRUNNION PLUG PLATE
44	2	SA 240 304	11 GAGE (1/8" THK.) SHEET.	POCKET TRUNNION PLUG PLATE
45	2	SA 240 304	11 GAGE (1/8" THK.) SHEET.	POCKET TRUNNION PLUG PLATE
46	2	SA 240 304	11 GAGE (1/8" THK.) SHEET.	POCKET TRUNNION PLUG PLATE
47	2	SA 240 304	11 GAGE (1/8" THK.) SHEET.	POCKET TRUNNION PLUG PLATE
48	4	SA-193 GRADE B7	3/8 - 16 UNC SCREW	POCKET TRUNNION PLUG SCREW
49	54	S/S	11 GAGE (1/8" THK.) SHEET.	CLOSURE BOLT WASHER
50	40	SA-193-B7	1 3/4"-5UNC SOCKET SET SCREW	TOP FLG. LIP HOLE PLUGS
51	20	SA-193-B7	1"-8UNC SOCKET SET SCREW	TOP FLG. SIDE HOLE PLUGS
52	16	SA-193-B7	1 3/4"-8UNC SOCKET SET SCREW	BOTTOM PLATE HOLE PLUGS
53	---	---	DELETED	---
54	4	SA-193-B7	1/2-13UNC SOCKET SET SCREW	THREADED PLUG

NOTES:
1) ALL DIMENSIONS ARE APPROXIMATE.
2) HOLTITE IS A NEUTRON SHIELD MATERIAL WITH NOMINALLY 1 WT. % B_4C, 6 WT. % H, AND A DENSITY OF 1.68g/cm^3.
3) ITEMS 12 THRU 16, MATERIAL SA-516-GR 70 IS TO BE NORMALIZED.
4) THICKNESS OF ITEM 16 MAY VARY DEPENDING ON THICKNESSES OF ITEMS 12-15.
5) ITEMS 2,12-17 MAY BE MADE FROM MORE THAN ONE PIECE.

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-C1478)

REF. DWG. C1395 & C1396.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-3-99 REVISED AS INDICATED	11/19/99	B.G. 11/19/99	MS 11/19/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	2	ALLOY "X" SEE NOTE 1.	PLATE 5/16" THK.	BASKET CELL PLATE
1B	1		PLATE 5/16" THK	BASKET CELL PLATE
1C	2		PLATE 5/16" THK	BASKET CELL PLATE
1D	1		PLATE 5/16" THK	BASKET CELL PLATE
1E	1		PLATE 5/16" THK	BASKET CELL PLATE
1F	22		PLATE 5/16" THK	BASKET CELL PLATE
1G	1		PLATE 5/16" THK	BASKET CELL PLATE
1H	2		PLATE 5/16" THK .	BASKET CELL PLATE
2	24		PIPE 3"-SCH 80 LGTH AS REQD.	UPPER FUEL SPACER PIPE
3A (3B)	84(12)	BORAL	.075"THK.	NEUTRON ABSORBER
4A (4B)	84(12)	ALLOY "X" SEE NOTE 1.	.06" THK. SHEATHING	SHEATHING
5A	4		PLATE 5/16"THK	BASKET CELL PLATE
5B	4		PLATE 5/16"THK	BASKET CELL PLATE
5C	4		PLATE 1.5" APP. THK.	BASKET SUPPORT
5D	4		2 1/2" W	BASKET SUPPORT
5E	4		2 "WIDE X THICKNESS AS REQD.	BASKET SUPPORT
5F	-		DELETED	---
5G	4		1 1/4" W X THICKNESS AS REQUIRED	BASKET SUPPORT SHIM
5H	---		DELETED	-----
6	1		1/2" THK CYLINDER.	SHELL
7	1		BASEPLATE 2 1/2" THK	BASEPLATE
8A	22		9/32"THK. ANGLE	BASKET CELL ANGLE
8B	2		9/32"THK. CHANNEL	BASKET CELL CHANNEL
9A	1		5/16" THK.PLATE	BASKET SUPPORT
9B	2		5/16"THK. PLATE	BASKET SUPPORT
9C	1		5/16"THK. PLATE	BASKET SUPPORT
9D	AS REQD		AS REQUIRED	BASKET SUPPORT
9E	----		DELETED	---
9F	---		DELETED	---
9G	---		DELETED	---
9H	---		DELETED	---
10	4		PLATE 3/4" THK.	LIFT LUG

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-C1478)

REF. DWG. C1395 & C1396.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-3-99 REVISED AS INDICATED	11/22/99	B.G. 11/22/95	11/22/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
11	4	ALLOY "X" SEE NOTE 1.	PLATE 3/4" THK.	LIFT LUG BASEPLATE
12	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD.	DRAIN SHIELD BLOCK
13A	2	304 S/S	BAR 2 11/16" OD	VENT AND DRAIN TUBE
13B	2	304 S/S	BAR 2 1/4 OD	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	9 1/2" THK.	MPC LID
15	1	↓	RING 3/8" THK.	MPC CLOSURE RING
16	---	----	DELETED	---
17	----	----	DELETED	----
18	AS REQD	ALLOY "X" SEE NOTE 1.	AS REQUIRED	BASKET SUPPORT SHIM
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK.	PORT COVER PLATE
20	24	A-193-B8 OR SIMILAR	3/4"-10UNC LG. BOLT	UPPER FUEL SPACER BOLT
21	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" X 2" X THICKNESS AS REQUIRED	LIFT LUG SHIM
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC SCREW	LID LIFT HOLE PLUG
24	24	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1.	LENGTH, WIDTH AND THICKNESS OF SHIMS AS REQUIRED.	LID SHIM
26	1	S/S	COUPLING	COUPLING
27	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" PLATE	UPPER FUEL SPACER END PLATE
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD.	VENT QUICK DISCONN. CPLG.
29	4	ALLOY "X" SEE NOTE 1.	BAR 3/4" OD.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173 1/2"APPROX. LG.	DRAIN LINE
31	--	-------	DELETED	-----
32	24		6" SQ. TUBING X 1/4" WALL LENGTH AS REQ'D.	LOWER FUEL SPACER CLUMN
33A	24		PLATE 3/8" THK	LOWER FUEL SPCAER END PLATE
33B	24	↓	PLATE 3/8" THK	LOWER FUEL SPCAER END PLATE
35	AS REQ'D.	ALUM. ALLOY 1100 & S/S	1/8" THICK X 176 1/2" LG. ALUM. SHEET WITH S/S SPRINGS	HEAT CONDUCTION ELEMENTS
36	2	ALUMINUM	0.065" THK SHT	SEAL WASHER
37	2	S/S	1/4" DIA BAR	SEAL WASHER BOLT
38	2	ALLOY "X" SEE NOTE 1.	1/8"THK	DRAINLINE
39	8	ALLOY "X" SEE NOTE 1.	1/8" THK	DRAINLINE

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. MINIMUM BORAL B-10 LOADING IS 0.0267 g/cm^2. BORAL TO BE PASSIVATED PRIOR TO INSTALLATION.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5C,5D,5E,5G 9A,9B,9C,9D,16,18,34 AND 35 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-C1479)

REF. DWGS. C1401 & C1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-3-99 REVISED AS INDICATED	[signature] 11/22/99	[signature] B.G. 11/22/99	[signature] 11/22/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	3	ALLOY "X" SEE NOTE 1.	PLATE 1/4" THK.	BASKET CELL PLATE
1B	4	↓	PLATE 1/4" THK.	BASKET CELL PLATE
1C	2		PLATE 1/4" THK.	BASKET CELL PLATE
1D	2		PLATE 1/4" THK.	BASKET CELL PLATE
1E	78		PLATE 1/4" THK.	BASKET CELL PLATE
2	68	▽	3"- SCH 80 PIPE LGTH AS REQD.	UPPER FUEL SPCER CLUMN
3A	116	BORAL	.101"THK. .	NEUTRON ABSORBER
4A	116	ALLOY "X" SEE NOTE 1.	.075" THK. SHEATHING	SHEATHING
5	8	↓	BAR 1" WIDE X 168" LG X THICKNESS AS REQUIRED	BASKET SUPPORT SHIM
6	1		1/2" THK PLATE	SHELL
7	1		BASEPLATE 2 1/2" THK	BASEPLATE
8	8		PLATE 5/16"THK.	BASKET SUPPORT
9A	4		BAR 1" W.	BASKET SUPPORT
9B	---		DELETED	---
9C	8		2 1/2" WIDE PLATE	BASKET SUPPORT
9D	AS REQD		AS REQUIRED	BASKET SUPPORT
10	4		PLATE 3/4" THK.	LIFT LUG
11	4		PLATE 3/4" THK. .	LIFT LUG BASEPLATE
12	1	▽	BAR 3.75"OD.	DRAIN SHIELD BLOCK
13A	2	304 S/S	BAR 2 11/16" OD	VENT AND DRAIN TUBE
13B	2	304 S/S	BAR 2 1/4" OD	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	10" THK. 10" THK.	MPC LID
15	1	ALLOY "X" SEE NOTE 1.	RING 3/8" THK. RING 3/8" THK.	MPC CLOSURE RING
16	---	---	DELETED	---

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BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-C1479)

REF. DWGS. C1401 & C1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S. GEE 11-3-99 REVISED AS INDICATED	11/15/99	B.G. 11/19/99	MS 11/19/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
17	1	ALLOY "X" SEE NOTE 1.	1" THK PLATE	SHELL (MPC-68F)
18	AS REQD	ALLOY "X" SEE NOTE 1.	AS REQUIRED	BASKET SUPPORT
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK	PORT COVER PLATE
20	68	A-193-B8 OR SIMILAR	3/4"-10UNC HEX BOLT	UPPER FUEL SPACER BOLT
21	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" W X 2' LG X THICKNESS AS REQUIRED	LIFT LUG SHIM
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC SOCKET SET SCREW.	LIFT HOLE PLUG
24	68	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1	LENGTH, WIDTH , THICKNESS AND QUANTITY AS REQD.	LID SHIM
26	1	S/S	COUPLING	COUPLING
27			DELETED	
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD.	VENT SHIELD BLOCK
29	4	ALLOY "X" SEE NOTE 1.	BAR .75"OD	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173" APPROX.LG.	DRAIN LINE
31	--	--	DELETED	---
32	--	--	DELETD	---
33	68	ALLOY "X" SEE NOTE 1.	4" SQ. TUBE X 1/4" WALL LENGTH AS REQD. (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER CLUMN
34A	68	ALLOY "X" SEE NOTE 1.	3/8" THK. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
34B	68	ALLOY "X" SEE NOTE 1.	3/8" THK. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
35	-------	-------------	DELETED	
36	-------	-------------	DELETED	
37	AS REQD	ALUM. ALLOY 1100 & S/S	1/8" THK. ALUM. SHEET W/S/S SPRINGS.	HEAT CONDUCTION ELEMENTS
38	2	ALUMINUM	.065" THK SHT	SEAL WASHER
39	2	S/S	1/4" DIA	SEAL WASHER BOLT
40	2	ALLOY "X" SEE NOTE 1	1/8" THK. SHEET	DRAIN LINE
41	8	ALLOY "X" SEE NOTE 1.	1/8" THK. SHEET	DRAIN LINE

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. BORAL B-10 LOADING IS 0.0372 g/cm . BORAL TO BE PASSIVATED PRIOR TO INSTALLATION. FOR MPC-68F, MINIMUM BORAL B-10 LOADING IS 0.01 g/cm^2.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5,8,9A,9C,9D,16,18, AND 37 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

BILL OF MATERIALS HI-STAR 100 SYSTEM FAILED FUEL CANISTER (BM-C1819)

REF. DWG. C1783 & C1784

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	S.GEE 11-19-99 ISSUED FOR APPROVAL	11/22/99	B.G. 11/22/99	11/22/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1	1	SA430-WP304	CONCENTRIC REDUCER	LEAD IN
2	1	SA479-304	ROUND BAR	LEAD IN COLLAR
3	1	SA479-304	ROUND BAR	LOCK PIN
4	1	SA240-304	SHEET	ENGAGEMENT PIN
5	2	SA479-304	ROUND BAR	ENGAGEMENT PIN
6	4	304 SST.	MESH	WIRE MESH
7	2	SA479-304 (or) SA240-304	3/16 BAR	RIM BAR (LONG)
8	2	SA479-304 (or) SA240-304	3/16 BAR	RIM BAR (SHORT)
9	2	SA479-304 (or) SA240-304	3/16 BAR	SIDE (SHORT)
10	2	SA479-304 (or) SA240-304	3/16 BAR	SIDE (LONG)
11	2	SA479-304 (or) SA240-304	1/2 SQ. BAR	LOAD TAB
12	1	SA479-304	Ø 1-1/2 ROUND BAR	LOAD TAB HUB
13	1	SA479-304	Ø 1-1/2 ROUND BAR	LOCKING SHAFT
14	1	SA479-304	Ø 1/8 ROUND BAR	LOCKING PIN
15	1	SA479-304	Ø 3/8 ROUND BAR	SHEAR PIN
16	1	304 SST. (or) 316 SST.	COMPRESSION SPRING	LOCKING SPRING
17	1	SA240-304	3/16 PLATE	CLOSURE FRAME BASE PLATE
18	2	304 SST.	1/16 COTTER PIN	COTTER PIN
19	1	SA240-304	3/8 PLATE	FUEL SPACER TOP PLATE
20	1	304 SST.	1/4 WALL TUBING	FUEL SPACER TUBING
21	1	304 SST.	MESH	WIRE MESH
22	2	SA240-304	11 GA. SHEET	CANISTER SLEEVE
23	1	SA240-304	11 GA. SHEET	CANISTER COLLAR
24	1	SA240-304	11 GA. SHEET	CANISTER BOTTOM
25	1	SA240-304	PLUG	LEAD-IN CAP
26	1	SA240-304	11 GA. SHEET	SCREEN RING
27	4	SA240-304	11 GA. SHEET	SCREEN RING
28	1	SA312-304	PIPE	LEAD-IN EXTENSION

NOTE: DIMENSIONS ARE APPROXIMATE

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1396	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1395 SHT 1 OF 4

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1396	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1395 SHT 2 OF 4

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B		
CLIENT N/A		
COMPANION DRAWINGS C1396		REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1395 SHT 3 OF 4	

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1396	REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1395 SHT 4 OF 4

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1396 SHT 1 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1396 SHT 2 OF 6

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1396 SHT 3 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1396 SHT 4 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 0
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1396 SHT 5 OF 6

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1395	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1396 SHT 6 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION CROSS SECTIONAL VIEW OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 1 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF TOP FLANGE & BOTTOM PLATE OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 2 OF 7

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF BOLT HOLE & BOLT OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 3 OF 7

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF CLOSURE PLATE TEST PORT AND NAME PLATE DETAIL OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 4 OF 7

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FEB 17-99	0	FOR APPROVAL	C.Y. 2-17-99	B.G.U. 2-20-99	B.G. 2-20-99	V.G. 2-20-99
DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	ENG.	Q. A.

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
DETAIL OF LIFTING TRUNNION & LOCKING PAD
OF HI-STAR 100 OVERPACK
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1398, C1399

REV. 0

PROJECT No. 5014

P.O. No. N/A

DRAWING No. C1397 SHT. 5 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF SHEAR RING AND CLOSURE PLATE BOLT INSTALLATION OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 6 OF 7

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF SHEAR RING OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1398, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1397 SHT. 7 OF 7

2 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 OVERPACK ORIENTATION CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1397, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1398 SHT. 1 OF 3

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION DETAIL OF DRAIN & RUPTURE DISK OF HI-STAR 100 OVERPACK CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS C1397, C1399	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1398 SHT. 2 OF 3

A

2 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
DETAIL OF VENT & PORT PLUG
OF HI-STAR 100 OVERPACK
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS
C1397, C1399

REV.
1

PROJECT No. 5014

P.O. No. N/A

DRAWING No.
C1398 SHT. 3 OF 3

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q.A.
NOV. 5-99	1	REVISED AS SHOWN	S. GEE 11-5-99	[illegible]	[illegible]	[illegible]
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.U.. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
SECTION "G" - "G"
OF HI-STAR 100 OVERPACK
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1397, C1398

REV. 1

PROJECT No. 5014

P.O. No. N/A

DRAWING No. C1399 SHT. 1 OF 3

B

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NOV. 5-99	1	REVISED AS SHOWN	S. GEE 11-5-99	[illegible]	[illegible]	[illegible]
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.U. 2-20-99	B.G. 2-20-99	V.G. 2-20-99
DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
SECTION "X"-"X" & VIEW "Y"
OF HI-STAR 100 OVERPACK
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1397, C1398

REV. 1

PROJECT No. 5014

P.O. No. N/A

DRAWING No. C1399 SHT. 2 OF 3

B

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV. 5-99	1	REVISED AS SHOWN	S.GEE 11-5-99	[illegible]	[illegible]	[illegible]
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.U. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN

ANALYSIS

CONSULTING

DESCRIPTION
DETAIL OF TRUNNION POCKET FORGING
OF HI-STAR 100 OVERPACK
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS
C1397, C1398

REV. 1

PROJECT No. 5014

P.O. No. N/A

DRAWING No.
C1399 SHT. 3 OF 3

B

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	REVISED AS SHOWN	S.GEE 11-6-99	[illegible]	B.G. [illegible]	[illegible] 11/19/99
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.U. 2-20-99	B.G. 2-20-99	V.G.. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1402 | REV. 1

PROJECT No. 5014
P.O. No. N/A
DRAWING No. C1401 SHT 1 OF 4

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	REVISED AS SHOWN	S.GEE 11-6-99	[illegible] 11/20/99	[illegible]	[illegible]
FEB. 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.B. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS	REV.
C1402	1

PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1401 SHT. 2 OF 4

B

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	ENG.	Q. A.
NOV. 18-99	1	FOR REVISION	J.A. 11-18-99	[illegible]	B.G. [illegible]	[illegible]
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1402 | REV. 1

PROJECT No. 5014 | DRAWING No. C1401 SHT 3 OF 4

P.O. No. N/A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	REVISED AS SHOWN	S.FEE 11-6-99	11/20/99	B.G. 11/20/99	11/20/99
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G.D. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1402 | REV. 1

PROJECT No. 5014
P.O. No. N/A
DRAWING No. C1401 SHT 4 OF 4

B

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	[illegible]	S.GEE 11-6-99	[illegible]	[illegible]	[illegible]
FEB 19-99	0	FOR APPROVAL	C.Y. 2-10-99	B.G.B. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC NO. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1401 — REV. 1

PROJECT No. 5014
P.O. No. N/A
DRAWING No. C1402 SHT 1 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	REVISED AS SHOWN	S.BEE 11-6-99			
FEB 19-99	0	FOR APPROVAL	C.Y. 2-19-99	B.G.B. 2-20-99	B.G. 2-20-99	V.G. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS	REV.
C1401	1

PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1402 SHT 2 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	ENG.	Q. A.
NOV. 18-99	1	FOR REVISION	J.A. 11-18-99	11/19/99	B.G. 11/19/99	11/19/99
FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G. 2-18-99	B.G. 2-18-99	V.G. 2-18-99

REVISION

HOLTEC INTERNATIONAL — EQUIPMENT DESIGN, ANALYSIS, CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS	REV.
C1401	1

PROJECT No. 5014	DRAWING No.
P.O. No. N/A	C1402 SHT 3 OF 6

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS - CONSULTING —	
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION CoC NO. 9261, APPENDIX B		
CLIENT N/A		
COMPANION DRAWINGS C1401		REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1402 SHT 4 OF 6	

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FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FEB 18-99	0	FOR APPROVAL	C.Y. 2-18-99	B.G. 2-18-99	B.G. 2-18-99	V.G. 2-18-99
DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	ENG.	Q. A.

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC NO. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1401 | REV. 0

PROJECT No. 5014 | DRAWING No. C1402 SHT 5 OF 6

P.O. No. N/A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

DATE	REV.	DESCRIPTION	PREP. BY:	CHECKED BY:	P.M.	Q. A.
NOV 6-99	1	REVISED AS SHOWN	S. GEE 11-6-99	[illegible] 11/20/99	[illegible] 11/20/99	[illegible] 11/20/99
FEB 19-99	0	FOR APPROVAL	C.T. 2-18-99	B.G.B. 2-20-99	B.G. 2-20-99	V.E. 2-20-99

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS •
CONSULTING —

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION
CoC No. 9261, APPENDIX B

CLIENT N/A

COMPANION DRAWINGS C1401 | REV. 1

PROJECT No. 5014 | DRAWING No. C1402 SHT 6 OF 6

P.O. No. N/A

FIGURE WITHHELD AS
SENSITIVE UNCLASSIFIED
INFORMATION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION HI-STAR 100 IMPACT LIMITER CoC No. 9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 1
PROJECT No.5014	DRAWING No.
P.O. No. VARIOUS	C1765 SHT. 1 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL		EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 BOTTOM IMPACT LIMITER CoC No. 9261, APPENDIX B		
CLIENT VARIOUS		
COMPANION DRAWINGS NONE		REV. 1
PROJECT No. 5014	DRAWING No. C1765 SHT. 2 OF 7	
P.O. No. VARIOUS		

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 TOP IMPACT LIMITER CoC No. 9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 1
PROJECT No. 5014	DRAWING No. C1765 SHT. 3 OF 7
P.O. No. VARIOUS	

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 TOP IMPACT LIMITER CoC No.9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 1
PROJECT No.5014 P.O. No. VARIOUS	DRAWING No. C1765 SHT. 4 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

CONSENT OF HOLTEC INTERNATIONAL.

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 TOP IMPACT LIMITER DETAIL OF ITEM #6 CoC No. 9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 1
PROJECT No.5014	DRAWING No.
P.O. No. VARIOUS	C1765 SHT. 5 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 IMPACT LIMITER HONEYCOMB DETAILS CoC No. 9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 1
PROJECT No. 5014	DRAWING No.
P.O. No. VARIOUS	C1765 SHT. 6 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

CONSENT OF HOLTEC INTERNATIONAL.

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION HI-STAR 100 BOTTOM IMPACT LIMITER CoC No. 9261, APPENDIX B	
CLIENT VARIOUS	
COMPANION DRAWINGS NONE	REV. 0
PROJECT No. 5014	DRAWING No.
P.O. No. VARIOUS	C1765 SHT. 7 OF 7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 ASSEMBLY FOR TRANSPORT CoC No. 9261, APPENDIX B	
CLIENT N/A	
COMPANION DRAWINGS ---	REV. 1
PROJECT No. 5014 P.O. No. N/A	DRAWING No. C1782

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS - CONSULTING	
DESCRIPTION: GENERAL ARRANGEMENT DAMAGED FUEL CONTAINER CoC No. 9261, APPENDIX B		
CLIENT: LATER		
COMPANION DRAWINGS: C1784		
PROJECT No.: 5014	DRAWING No.:	REV.:
P.O. No.: LATER	C1783	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION: DAMAGED FUEL CONTAINER DETAILS CoC No. 9261, APPENDIX B	
CLIENT: LATER	
COMPANION DRAWINGS: C1783	PREP. BY: C.Y. 2-23-09
PROJECT No.: 5014 SCALE: FULL P.O. No.: LATER	DRAWING No.: C1784 REV.: 1 SHEET: 1 of 1

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FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [illegible]	9/2/83	D-1 CANISTER ASSEMBLY
EDO CK'D. BY	24 AUG 83	
DWN. BY PGS	05 AUG 83	

SCALE	SIZE	DWG. NO.	REV.
NONE	B	9317.1-120-2	0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

APPROVALS	DATE	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.			
PROJ. GTT/KG	9/2/83	D-1 CANISTER LID ASSEMBLY			
Q/A	9/2/83				
CK'D. BY EDO	2 SEPT 83				
DWN. BY PGS	02 SEP 83	SCALE 1/1	SIZE B	DWG. NO. 9317.1-120-3	REV. 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
		TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
PROJ. GTT/KL	9/2/83	
Q/A	9/2/83	D-1 CANISTER BODY
CK'D. BY EDO		
DWN. BY PGS	02 SEP 93	

SCALE	SIZE	DWG. NO.	REV.
NONE	B	9317.1-120-4	0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	03 OCT 83	SEE DCN 9317.1-120-5	PGS	EDO			

APPROVALS	DATE
PROJ. GTT/KG	9/2/83
Q/A	9/2/83
EDO CK'D. BY	
Z. DURAS DWN. BY PGS	02 SEP. 83

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

D-1 CANISTER
BOTTOM ASSEMBLY

SCALE	SIZE	DWG. NO.	REV
1/2	B	9317.1-120-5	1

STANPAT PRODUCTS INC. PORT WASHINGTON,

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	11 OCT 83	SEE DCN 9317.1-120-7	PGS	EDO		[illegible]	KG

APPROVALS	DATE
PROJ. GTT/KG	9/2/83
Q/A [illegible]	9/2/83
EDO CK'D. BY	24 AUG 83
Z. DURAS DWN. BY	8-5-83

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

D-1 CANISTER
LOWER LID BOX ASSY

SCALE	SIZE	DWG. NO.	REV.
1/1	B	9317.1-120-6	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
PROJ. GTT/KG	9/2/83	D-1 CANISTER BUMPER PLATE
Q/A	9/2/83	
EDO CK'D. BY	24 AUG 83	
Z. DURAS DWN. BY	8-5-83	1/1 SCALE — B SIZE — 9317.1-120-7 DWG. NO. — 0 REV.

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [signature]	9/6/83	D-1 CANISTER BALE
CK'D. BY EDO	9/2/83	
DWN. BY PRB	02 SEP 83	SCALE: 1 / SIZE: B / DWG. NO.: 9317.1-120-8 / REV.: 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	9/83	SEE DCN 9317.1-120-8	PGB	EDO			KG

APPROVALS	DATE
PROJ. GTT/KG	9/2/83
Q/A	9/2/83
EDO CK'D. BY	24 AUG 83
Z. DURAS DWN. BY	8-5-83

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

D-1 CANISTER HANGER

SCALE	SIZE	DWG. NO.	REV.
1/1	B	9317.1-120-9	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	11/85	SEE DCN 9317.1-120-9	PGB	EDO		[illegible]	KG

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [illegible]	9/6/83	D-1 CANISTER LID BOX -
CK'D. BY EDO	24 AUG 83	
DWN. BY PGB	05 AUG 83	SCALE 1/2 · SIZE B · DWG. NO. 9317.1-120-10 · REV. 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	03 OCT 83	SEE DCN 9317.1-120-4	PGB	EDO		[illegible]	[illegible]

APPROVALS	DATE
PROJ. GTT/KL	9/2/83
Q/A [illegible]	9/2/83
CK'D. BY EDO	9/2/83
DWN. BY PGB	02 SEP 83

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

D-1 CANISTER
LID FRAME

SCALE	SIZE	DWG. NO.	REV
1/1	B	9317.1-120-11	1

STANPAT PRODUCTS INC. PORT WASHINGTON.

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [signature]	9/2/83	D-1 CANISTER GUIDE BAR
CK'D. BY E DO	24 AUG 83	
DWN. BY Z. DURAS	7-29-83	SCALE 1/1 · SIZE B · DWG. NO. 9317.1-120-13 · REV. 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	9/2/83	D-1 CANISTER FUEL SUPPORT PLATE
EDO CK'D. BY	9/2/83	
Z. DURAS DWN. BY	02 SEP 83	

SCALE	SIZE	DWG. NO.	REV.
1/1	B	9317.1-120-14	0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KE	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	9/2/83	D-1 CANISTER SCREEN SUPPORT PLATE
CK'D. BY EDO	24 AUG 83	
DWN. BY PGB	18 AUG 83	SCALE 1/1 · SIZE B · DWG. NO. 9317.1-120-15 · REV. 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KL	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	9/2/83	D-1 CANISTER STRUT
CK'D. BY EDO	24 AUG 83	
DWN. BY PGB	18 AUG 83	SCALE 1/1 · SIZE B · DWG. NO. 9317.1-120-17 · REV. 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	OCT 83	SEE DCN 9317.1-120-10	PGS	EDO		[illegible]	KG

APPROVALS	DATE
PROJ. GTT/KG	9/6/83
Q/A [illegible]	9/6/83
CK'D. BY EDO	6 SEPT 83
DWN. BY Buchholtz	6 SEPT. '83

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

D-1 CANISTER SCREEN

SCALE	SIZE	DWG. NO.	REV.
1/1	B	9317-120-18	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

2	9/7/84	SEE DCN 9317.1-120-38	AG	[illegible]		[illegible]	[illegible]
1	OCT 83	SEE DCN 9317.1-120-11	PGB	EDO		[illegible]	KG.
NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [illegible]	9/6/83	D-1 CANISTER THIN (INNER) WIPER
CK'D. BY EDO	24 AUG 83	
Buchholtz DWN. BY	24 AUG. '83	

SCALE	SIZE	DWG. NO.	REV.
1/1	B	9317.1-120-19	2

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ. GTT/KG	9/2/83	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	9/2/83	D-1 CANISTER SPACER
EDO CK'D. BY	9/2/83	
DWN. BY PCB	02 SEP 83	

SCALE	SIZE	DWG. NO.	REV.
1/2	B	9317.1-120-20	C

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ.	2/17/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A		D-2/3 FUEL SPACER FOR TN-9 CASK
CK'D. BY EDO	17 FEB 84	
DWN. BY PG3	20 OCT 83	SCALE: 1/2 · SIZE: B · DWG. NO.: 9317.1-120-21 · REV.: 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.

APPROVALS	DATE	
PROJ.	9/6/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	7/6/84	D-1 CANISTER THICK WIPER
CK'D. BY	9/6/84	
DWN. BY AG	5 SEPT 84	SCALE: 1/1 · SIZE: B · DWG. NO.: 9317.1-120-22 · REV.: 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

APPROVALS	DATE	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
PROJ.	9/4/84	
Q/A	9/6/84	D-1 CANISTER LID ASSEMBLY FOR FAILED FUEL
CK'D. BY	9/4/84	
DWN. BY AG	5 SEP 84	SCALE: — / SIZE: B / DWG. NO.: 9317.1-120-23 / REV.: 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	02 NOV 84	SEE DCN 9317.1-182-5	PB	EDO			KL

APPROVALS	DATE	
PROJ.	8/14/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	8/14/84	THORIA ROD CANISTER ASSEMBLY
CK'D. BY EDO	9 JULY 84	
A.GALASSO DWN. BY	2 JULY 84	NONE SCALE · B SIZE · 9317.1-182-1 DWG NO. · 1 REV.

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	8/28/84	SEE DCN 9317.1-182-2	AG	EDO			

APPROVALS	DATE	
PROJ.	8/11/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	8/14/84	THORIA ROD CANISTER SPACER
CK'D. BY EDO	9 JULY 84	
A. GALASSO DWN. BY	9 JULY 84	SCALE: FULL; SIZE: B; DWG. NO.: 9317.1-182-2; REV.: 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	8/24/84	SEE DCN 9317.1-182-1	AG	EDO			

APPROVALS	DATE
PROJ.	8/14/84
Q/A	8/14/84
CK'D. BY EDO	9 JULY 84
A.GALASSO DWN. BY	5 JULY 84

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

THORIA ROD CANISTER
SEPARATOR PLATE ASSEMBLY

SCALE	SIZE	DWG. NO.	REV.
NONE	B	9317.1-182-3	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	02 NOV 84	SEE DCN 9317.1-182-6	PSB	EDO		[illegible]	K64-6T

APPROVALS	DATE	
PROJ. [illegible]	8/14/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A [illegible]	8/14/84	THORIA ROD CANISTER RETAINING PLATE ASSEMBLY
CK'D. BY EDO	9 JULY 84	
DWN. BY A.GALASSO	5 JULY 84	

SCALE	SIZE	DWG. NO.	REV.
NONE	B	9317.1-182-4	1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A	PROJ.
1	8/28/84	SEE DCN 9317.1-182-3	AG.	EDO			

APPROVALS	DATE	
PROJ.	8/17/84	TRANSNUCLEAR, INC. WHITE PLAINS, N.Y.
Q/A	8/14/84	THORIA ROD CANISTER RETAINING PLATE DETAILS
EDO CK'D. BY	9 JULY 84	
A.GALASSO DWN. BY	3 JULY 84	SCALE: NONE; SIZE: B; DWG. NO.: 9317.1-182-5; REV.: 1

PARTS LIST				
ITEM NO	QTY	TITLE	DESCRIPTION	MATL
1	4	SEP PLATE ASSY	9317.1-182-3 REV 1	12 GA ASTM 240
2	1	SPACER	9317.1-182-2 REV 1	ASTM 240
3	1	RETAINING PLATE ASSY	9317.1-182-4 REV 1	12 GA ASTM 240
3.1	1	RETAINING PLATE	9317.1-182-5 REV 1	12 GA ASTM 240
3.2	4	RETAINING PLATE CLIP	9317.1-182-5 REV 1	12 GA ASTM 240
3.3	4	SCREW	#10-32UNC-2A FLAT HEAD SLOTTED X.62 LG	3xx SS
3.4	4	LOCKWASHER	#10	3xx SS
3.5	4	NUT	#10-32 UNC-2B HEX HEAD	3xx SS
3.6	1	BALE	9317.1-182-7, REV. 0	(SS TP 304)

NO.	DATE	REVISIONS	DWN.	CK'D.		Q/A.	PROJ.
2	02 NOV 84	SEE DCN 9317.1-182-7	PGS	EDO			KGF
~~1~~	~~8/28/84~~	~~SEE DCN 9317.1-182-4~~	~~AG~~	~~EDO~~			

APPROVALS	DATE
PROJ	8/14/84
Q/A	8/14/84
CK'D. BY	8/14/84
A. GALASSO DWN. BY	9 AUG 84

TRANSNUCLEAR, INC.
WHITE PLAINS, N.Y.

THORIA ROD CANISTER
PARTS LIST

SCALE	SIZE	DWG. NO.	REV.
NONE	B	9317.1-182-6	2

This report has been prepared in the format and content suggested in NRC Regulatory Guide 7.9 [1.0.3]. The purpose of this chapter is to provide a general description of the design features and transport capabilities of the HI-STAR 100 packaging including its intended use. This chapter provides a summary description of the packaging, operational features, and contents, and provides reasonable assurance that the package will meet the regulations and operating objectives. Table 1.0.1 contains a listing of the terminology and notation used in preparing this licensing application report.

This SAR was prepared prior to the issuance of the draft version of NUREG-1617 [1.0.5]. To aid NRC review, additional tables and references have been added to facilitate the location of information needed to demonstrate compliance with 10CFR71 as outlined by NUREG-1617. Table 1.0.2 provides a matrix of the 10CFR71 requirements as outlined in NUREG-1617, the format requirements of Regulatory Guide 7.9, and reference to the applicable SAR section(s) that address(es) each topic.

The HI-STAR 100 Topical Safety Analysis Report (TSAR) [1.0.6] has been submitted to the NRC for a Certificate of Compliance for HI-STAR 100 to store spent nuclear fuel at an Independent Spent Fuel Storage Installation (ISFSI) facility under requirements of 10CFR72, Subpart L [1.0.4] (Docket Number 72-1008).

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.1.1 is the first figure in Section 1.1 of Chapter 1 (which is the next section in this chapter).

Revision of this document to Revision ~~8~~ *9* was made on a page level. Therefore, if any change occurs on a page or figure, the affected page or figure was updated to Revision ~~8~~ *9*.

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

AL-STAR™ is the trademark name of the HI-STAR 100 impact limiter.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

Boral™ means Boral manufactured by AAR Advanced Structures.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Containment System Boundary means the enclosure formed by the overpack inner shell welded to a bottom plate and top flange plus the bolted closure plate with dual seals and the vent and drain port plugs with seals. It is also called the primary containment boundary when used with the inner (secondary) containment boundary of the MPC-68F.

Containment System means the HI-STAR 100 overpack which forms the containment boundary of the packaging intended to contain the radioactive material during transport.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, *as determined by the review of records,* greater than pinhole leaks or hairline cracks, missing fuel rods that are not replaced 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 to Fuel Debris.

Damaged Fuel Container means a specially designed enclosure for damaged fuel assemblies or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross solid particulates. The Damaged Fuel Container (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC. *DFCs authorized for use in the HI-STAR 100 System are the Holtec design or the Transnulcear Dresden Unit 1 design as shown on the applicable design drawings in Section 1.5.*

Enclosure Vessel means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, and closure ring which provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister. See also Helium Retention Boundary.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Exclusive use means the sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must issue specific instructions, in writing, for maintenance of exclusive use shipment controls, and include them with the shipping paper information provided to the carrier by the consignor.

Fuel Basket means a honeycomb structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Debris is fuel with known or suspected defects, such as ruptured fuel rods, severed rods, or loose fuel pellets. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered to be Fuel Debris.

Helium Retention Boundary or Enclosure Vessel means the enclosure formed by the MPC shell welded to the baseplate, lid, welded port cover plates, and closure ring.

HI-STAR 100 MPC means the sealed spent nuclear fuel container which consists of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior design dimensions.

HI-STAR 100 overpack or overpack means the cask which receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the containment boundary for radioactive materials, gamma and neutron shielding, and a set each of lifting and pocket trunnions for handling.

HI-STAR 100 System or HI-STAR 100 Packaging consists of the HI-STAR 100 MPC sealed within the HI-STAR 100 overpack with impact limiters installed.

Holtite™ is a trade name denoting an approved neutron shield material for use in the HI-STAR 100 System. In this application, Holtite-A is the only approved neutron shield material.

Holtite™-A is a commercially available neutron shield material developed by Bisco, Inc., and currently sold under the trade name NS-4-FR. The neutron shield material is specified with a ~~minimum~~ *nominal* B_4C loading of 1 weight percent. An equivalent neutron shield material with equivalent neutron shielding properties and composition, but not sold under the trade name NS-4-FR, may be used.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Impact Limiter means a set of fully-enclosed energy absorbers which are attached to the top and bottom of the overpack during transport. The impact limiters are used to absorb kinetic energy resulting from normal and hypothetical accident drop conditions. The HI-STAR impact limiters are called AL-STAR.

Important to Safety (ITS) means a function or condition required to transport spent nuclear fuel safely; to prevent damage to spent nuclear fuel, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, transported, and retrieved without undue risk to the health and safety of the public.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, 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).

Maximum Normal Operating Pressure (MNOP) means the maximum gauge pressure that would develop in the containment system in a period of 1 year under the heat condition specified in 10CFR71.71(c)(1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

MGDS is an acronym for Mined Geological Depository System.

MPC Fuel Basket means the honeycombed composite cell structure utilized to maintain subcriticality of the spent nuclear fuel. The number and size of the storage cells depends on the type of spent nuclear fuel to be transported. Each MPC fuel basket has sheathing welded to the storage cell walls for retaining the Boral neutron absorber. Boral is a commercially-available thermal neutron poison material composed of boron carbide and aluminum.

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. MPC is an acronym for multi-purpose canister. The MPCs used as part of the HI-STAR 100 Packaging (Docket No. 71-9261) are identical to the MPCs evaluated in the HI-STAR 100 Storage (Docket No. 72-1008) and HI-STORM 100 Storage (72-1014) [1.0.7] applications.

maximum capacity with design basis SNF. The maximum gross transport weight of the HI-STAR 100 Packaging is to be marked on the regulatory label plate as shown on the Design Drawings in Section 1.4.

1.2.1.2 Materials of Construction, Dimensions and Fabrication

All materials utilized to construct the HI-STAR 100 System are ASME Code materials, except the neutron shield, neutron poison, heat conduction inserts, thermal expansion foam, seals, rupture disks, aluminum honeycomb, ~~and~~ pipe couplings *and other material classified as Not Important to Safety*. The specified materials of construction along with the detailed dimensions are provided in the Design Drawings and Bills-of-Material 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.

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 which 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* 203-1/8 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 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 a vent port which is sealed by a threaded port plug with a seal. The bottom plate has a drain port which is sealed 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 which 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 the enclosure shell panels form the overpack outer enclosure shell (Figure 1.2.2). Atop the outer enclosure shell, rupture disks are positioned in a recessed area. The rupture disks relieve internal pressure which may develop as a result of the fire accident and subsequent off-gasing 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. Appendix 1.C provides material information on the thermal expansion foam. Figure 1.2.2 contains a mid-plane cross section 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. The paints applied to the overpack exposed exterior and interior surfaces are specified on the design drawings; the material data on the paint is provided in Appendix 1.C. The inner cavity of the overpack is coated with a paint appropriate to its higher temperatures and the exterior of the overpack is coated with a paint 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. Pocket trunnions are welded to the lower side of the overpack to provide a pivoting axis for rotation. The pocket trunnions are located slightly off-center to ensure proper rotation direction of the overpack. As shown in Figure 1.1.4, the trunnions do not protrude beyond the cylindrical envelope of the overpack outer enclosure shell. This feature reduces the potential for a direct impact on a trunnion in the event of an overpack side impact.

1.2.1.2.2 Multi-Purpose Canisters

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 and cylindrical height of each MPC is fixed. However, the number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. As the MPCs are interchangeable, they correspondingly have identical exterior dimensions. The outer diameter of the MPC is *nominally* 68-3/8 inches and the length is *approximately* 190-1/2 inches. Figures 1.2.3-1.2.5 depict the cross sectional views of the different MPCs. Detailed Design Drawings of the MPCs are provided in Section 1.4. Key operational and design parameters for the MPCs are outlined in Tables 1.2.2 and 1.2.3.

The construction features of the PWR MPC-24 and the BWR MPC-68 are similar. However, the PWR MPC-24 canister in Figure 1.2.5, which is designed for highly enriched PWR fuel, differs in construction in one important aspect: the fuel cells are physically separated from one another by a flux trap between each cell for criticality control. All MPC baskets are formed from an array of plates welded to each other, such that a honeycomb structure is created which resembles a multiflanged, closed-section beam in its structural characteristics.

The MPC fuel basket is positioned and supported within the MPC shell by a basket support structure welded to the inside of the MPC shell. Between the periphery of the basket, the MPC shell, and the basket supports, heat conduction elements are installed. These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes ***and a design*** which allow ~~sufficient flexibility to enable~~ a snug fit in the confined spaces and ~~for~~ ease of installation. The heat conduction elements are installed along the full length of the MPC basket, ***except at the drain pipe location,*** to create a nonstructural thermal connection which facilitates heat transfer from the basket to shell. In their ~~installed~~ *operating* condition, the heat conduction elements will conform to and contact the MPC shell and basket walls.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit placement of the empty MPC into the overpack. 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 end 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 edge-welded to the MPC shell. This lid is equipped with vent and drain ports which are utilized to remove moisture and air from the MPC and backfill the MPC with a specified ~~mass~~ *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. Threaded insert plugs are installed to provide shielding when the threaded holes are not in use.

All MPCs are designed to handle intact and damaged SNF and fuel debris in damaged fuel containers. However, at this time only the MPC-68 is being licensed to handle intact and damaged SNF/fuel debris. The definition and design basis characteristics for damaged fuel, fuel debris and intact SNF are provided in Subsection 1.2.3.

To distinguish an MPC-68 which is to transport fuel assemblies classified as fuel debris in damaged fuel containers, the MPC shall be designated as an "MPC-68F." The MPC-68F design to transport fuel debris is similar to the MPC-68 design to transport intact fuel and damaged fuel. The sole differences in the MPC-68F is a reduction in the required ^{10}B areal density in the Boral for the transport of fuel debris, an increase in the thickness of the MPC shell in the MPC lid region, a corresponding decrease in the outer diameter of the MPC lid and closure ring, and an increase in the MPC lid to shell weld size. A reduction in the required ^{10}B areal density of the

Boral is possible due to limited enrichment permitted. The modification to the MPC closure design is specified to ensure that the MPC provides the inner containment boundary for the transportation of fuel assemblies classified as fuel debris. Therefore, an MPC-68 which is to transport fuel assemblies classified as fuel debris is designated with the unique nomenclature, MPC-68F.

Intact SNF can be placed directly into the MPC-68F. Damaged SNF and fuel debris must be placed into damaged fuel containers prior to insertion into the MPC-68F. Figure 1.2.10 provides a sketch of ~~the~~ *a* damaged fuel container and Section 1.4 provides ~~a~~ *the* detailed design drawing*s*.

The MPC-68F provides 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.

The HI-STAR MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and aluminum heat conduction elements). 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 which warrants further explanation.

Alloy X is a material which should be acceptable as a Mined Geological Depository System (MGDS) waste package and which 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 which 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 application) may be one of the following materials (only a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC):

future. Nevertheless, to add another layer of insurance, only 75% ^{10}B credit of the fixed neutron absorber is assumed in the criticality analysis.

1.2.1.4.2 Holtite™ 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.

In the current submittal, Holtite-A is the only approved neutron shield material which fulfills the aforementioned criteria. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is a commercially available neutron shield under the trade name NS-4-FR and is specified with a ~~minimum~~ *nominal* B_4C loading of 1 weight percent for the HI-STAR 100 System. Appendix 1.B provides the Holtite-A material properties. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

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

Density

The specific gravity of Holtite-A is 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 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% ~~(minimum)~~ 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 ~~minimum~~ *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. Technical papers provided in Appendix 1.B validate the neutron shield material's long-term stability within the design temperature and the material's ability to resist the effects of a fire accident. Holtite-A has been utilized in similar applications and has been licensed for use in a transportation cask under Docket No. 71-9235.

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 of the HI-STAR 100 packaging onto the transport vehicle by restricting the motion of the package. Securing the upper portion of the HI-STAR 100 packaging onto the transport vehicle is accommodated by the saddle of the transport frame interfacing with the shear ring on the overpack top flange. The shear ring is an integral part of the overpack top flange and is simply a raised area of the top flange. The transport frame is designed to meet the applicable requirements specified by the American Association of Railroads and to satisfy the requirements of 10CFR71.45(b)(3) the ultimate load capacity of the tie-downs shall be shown to be less than the corresponding ultimate load capacities of either the shear ring or pocket trunnion.

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. 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 Overpack (storage-only) to the HI-STAR Overpack, or vice versa. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

1.2.1.6 Heat Dissipation

The HI-STAR 100 System can safely transport SNF by maintaining the fuel cladding temperature below the design limits specified in Table 1.2.3. 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 MPC in this licensing application 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 the fuel loading operations. Table 1.2.3 specifies the helium fill ~~mass~~ *pressure* to be placed in the MPC internal cavity ~~as a function of the free space. As instructed in the operating procedures of Chapter 7, the water drained from the loaded MPC is measured to determine the free volume in the MPC internal cavity with fuel. As a fill pressure is highly dependent on temperature, and the decay heat coupled with the vacuum drying process elevates the MPC internal temperature, it is much more accurate to measure the mass placed 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 pool surface. As an ALARA measure, dose rates are measured on the top of the overpack and MPC prior to removal from the pool to check for activated debris on the top 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. Dose rates are measured to ensure that the dose rates are within expected values. The Automated Welding System (AWS) is installed. The MPC water level is lowered slightly and 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. The water level is raised to the top of the MPC and the weld is hydrostatically tested. 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 ~~water level is raised to the top of the MPC again and then 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. ~~The volume of water displaced from the MPC is measured to determine the free volume inside the MPC. This information is used to determine the helium backfill requirements for the MPC.~~

The Vacuum Drying System (VDS) is connected to the MPC and is used to remove all residual water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and VDS lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated.

Following this dryness test, the VDS is disconnected, the Helium Backfill System (HBS) is attached, and 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 final passes. 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 final welds are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in

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.

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, a particular 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] and [1.2.7] which 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. Table 1.2.12 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and decay heat generation. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. 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 the HI-STAR 100 System is intact zircaloy clad fuel assemblies with the characteristics listed in Table 1.2.13 or intact stainless steel clad fuel assemblies with the characteristics listed in Table 1.2.19. 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 specified in Table 1.2.19. Stainless steel clad fuel assemblies can only be transported in the MPC-24 and MPC-68.

Fuel assemblies with missing pins cannot be classified as intact fuel unless dummy fuel pins, which occupy ~~an~~ ~~equal~~ volume ~~as~~ *equal to or greater than* the original fuel pins, replace the missing pins prior to loading. Any intact fuel assembly which falls within the geometric, thermal, container), while the overpack provides the primary containment boundary.

The fuel characteristics specified in Table 1.2.11 for the Dresden 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.

1.2.3.4 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 1.2.13, 1.2.14, and 1.2.19. 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. In order to qualify for transport in the HI-STAR 100 MPC, the SNF must satisfy the physical parameters listed in Tables 1.2.13, 1.2.14, or 1.2.19.

1.2.3.5 Thermal Payload Parameters

The principal thermal design parameter for the fuel is the peak fuel cladding temperature limit, 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 highest decay heat for a given enrichment, burnup, and cooling time. This decay heat design basis fuel is listed in Table 1.2.12. Section 5.2 describes the method used to determine the design basis fuel assembly type and calculate the decay heat load.

As can be seen in Table 3.3.7, the acceptable normal condition fuel cladding temperature limit decreases with increased cooling time. Therefore, the allowable decay heat load per fuel assembly must correspondingly decrease with increased fuel assembly cooling time. For example, the maximum decay heat load for 5-year cooled zircaloy clad BWR fuel in the MPC-68 is 272W, but for 10-year cooled zircaloy clad BWR fuel, the decay heat load is limited to 245W. To ensure the allowable fuel cladding temperature limits are not exceeded, Figure 1.2.12 specifies the allowable decay heat per assembly versus cooling time for zircaloy clad fuel in each MPC type. Tables 1.2.14 and 1.2.19 provide the maximum heat generation for damaged zircaloy clad fuel assemblies and stainless steel clad fuel assemblies, respectively. Any damaged fuel assembly or fuel assembly classified as fuel debris with a decay heat load equal to or less than the maximum value specified in Table 1.2.14 is acceptable for loading into the HI-STAR 100 System. Due to the conservative thermal assessment and the long cooling time of the damaged and stainless steel clad fuel, a respectively. Table 1.2.20 provides the burnup and cooling time characteristics which satisfy the radiological source term requirements for intact zircaloy clad fuel in each MPC type.

Table 1.2.15 and Figures 1.2.13 and 1.2.14 provide the axial distribution for the radiological source term for PWR and BWR fuel assemblies 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 features a basket without flux traps. In the MPC-68 basket, there is one panel of neutron absorber between two adjacent fuel assemblies. The MPC-24 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. The maximum initial ^{235}U enrichment for the MPC-24 is specified by the fuel array type in Table 1.2.10.

The MPC-24 Boral ^{10}B areal density is specified at a minimum loading of 0.0267 g/cm^2. The 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-1536 [1.2.5] 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.

1.2.3.8 Summary of Design Criteria

An intact zircaloy 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, and 1.2.13.

b. Its initial enrichment is less than that indicated by Table 1.2.13 for the MPC in which it is intended to be transported.

c. The period from discharge is greater than or equal to the minimum cooling time listed in Table 1.2.20 for the given burnup and minimum enrichment.

d. The average burnup of the assembly is equal to or less than the burnup specified in Table 1.2.20 for the given cooling time and minimum enrichment.

A damaged fuel assembly shall be placed in a damaged fuel container and shall meet the characteristics specified in Table 1.2.14 for transport in the MPC-68. Fuel assemblies classified as fuel debris shall be placed in a damaged fuel container and shall meet the characteristics specified in Table 1.2.14 for transport in the MPC-68F.

Stainless steel clad fuel assemblies shall meet the characteristics specified in Table 1.2.19 for transport in the MPC-24 or MPC-68.

MOX BWR fuel assemblies shall meet the requirements of Tables 1.2.13 and 1.2.14 for intact and damaged fuel/fuel debris, respectively.

No control components in PWR fuel are to be included with the fuel assembly. Only zircaloy fuel channels may be transported with the BWR fuel assemblies 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.

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

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

PARAMETER	PWR	BWR
Unloaded MPC weight (lb)	See Table 1.2.1	See Table 1.2.1
Minimum neutron absorber ^{10}B loading (g/cm^2)	0.0267 (MPC-24)	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	 100 100 125	 100 100 125
Total heat load, max. (kW)	20.0 (MPC-24)	18.5 (MPC-68)
Maximum permissible peak fuel cladding temperature (°F)	See Table 3.3.7 (normal conditions) 1058° (accident conditions)	See Table 3.3.7 (normal conditions) 1058° (accident conditions)
MPC internal environment Helium filled (~~g-moles/ℓ of free space~~ *psig*)	~~0.1212 (MPC-24)~~ *≤22.2*	~~0.1218 (MPC-68)~~ *≤28.5*
MPC external environment/overpack internal environment Helium filled initial pressure (psig, at STP)	10	10
Maximum permissible reactivity including all uncertainties 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.

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 ~~(except 8x8 WE (QUAD+))~~	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	14x14 A	14x14 B	14x14 C	14x14 D	15x15 A
Clad Material (note 2)	Zr	Zr	Zr	SS	Zr
Design Initial U (kg/assy.)	≤ ~~402~~ *407*	≤ ~~402~~ *407*	≤ ~~410~~ *425*	≤ 400	≤ ~~420~~ *464*
Initial Enrichment (wt % ^{235}U)	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0	≤ 4.1
No. of Fuel Rods	179	179	176	180	204
Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.418
Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ ~~0.3840~~ *0.3880*	≤ 0.3890	≤ 0.3660
Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ ~~0.3770~~ *0.3805*	≤ 0.3835	≤ 0.3580
Fuel Rod Pitch (in.)	0.556	0.556	0.580	0.556	0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 150
No. of Guide Tubes	17	17	5 (see note 3)	16	21
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ ~~0.040~~ *0.0380*	≥ 0.0145	≥ 0.0165

Notes:
1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. Each guide tube replaces four fuel rods.
4. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array/Class	15x15 G	*15x15 H*	16x16 A	17x17A	17x17 B	17x17 C
Clad Material (note 2)	SS	*Zr*	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤420	*≤475*	≤~~430~~ *443*	≤~~450~~ *467*	≤~~464~~ *467*	≤~~460~~ *474*
Initial Enrichment (wt % ^{235}U)	≤4.0	*≤3.8*	≤4.6	≤4.0	≤4.0	≤4.0
No. of Fuel Rods	204	*208*	236	264	264	264
Clad O.D. (in.)	≥0.422	*≥0.414*	≥0.382	≥0.360	≥0.372	≥0.377
Clad I.D. (in.)	≤0.3890	*≤0.3700*	≤0.3320	≤0.3150	≤0.3310	≤0.3330
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 Tubes	21	*17*	5 (note 3)	25	25	25
Guide Tube Thickness (in.)	≥0.0145	*≥0.0140*	≥0.0400	≥0.016	≥0.014	≥0.020

Notes:

1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. Each guide tube replaces four fuel rods.
4. Description of the fuel assembly class designation is provided in Chapter 6.

Table 1.2.11
BWR FUEL ASSEMBLY CHARACTERISTICS (note 1)

Fuel Assembly Array/Class	6x6 A	6x6 B	6x6 C	7x7 A	7x7 B	8x8 A
Clad Material (note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ ~~108~~ *110*	≤ ~~108~~ *110*	≤ ~~108~~ *110*	≤ 100	≤ 195	≤ 120
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 3 for MOX rods.	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.0	≤ 4.0
No. of Fuel Rods	36	36 (up to 9 MOX rods)	36	49	49	64
Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Clad I.D. (in.)	≤ ~~0.4945~~ *0.5105*	≤ 0.4945	≤ 0.4990	≤ ~~0.4200~~ *0.4204*	≤ 0.4990	≤ 0.3620
Pellet Dia. (in.)	≤ ~~0.4940~~ *0.4980*	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ ~~0.4880~~ *0.4910*	≤ 0.3580
Fuel Rod Pitch (in.)	~~0.694~~ *0.710*	~~0.694~~ *0.710*	0.740	0.631	0.738	0.523
Active Fuel Length (in.)	≤ ~~110~~ *120*	≤ ~~110~~ *120*	≤ *77.5*	≤ ~~79~~ *80*	≤ 150	≤ ~~110~~ *120*
No. of Water Rods	*1 or* 0	*1 or* 0	0	0	0	*1 or* 0
Water Rod Thickness (in.)	~~N/A~~ *>0*	~~N/A~~ *>0*	N/A	N/A	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Notes:

1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. ≤ 0.612 wt. % ^{235}U and ≤ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu).
4. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array/Class	8x8 B	8x8 C	8x8 D	8x8 E	9x9 A	9x9 B
Clad Material (note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ 185	≤ 185	≤ 185	≤ ~~180~~ *185*	≤ ~~173~~ *177*	≤ ~~173~~ *177*
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	63 *or 64*	62	60 *or 61*	59	74/66 (note 3)	72
Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4400	≥ 0.4330
Clad I.D. (in.)	≤ ~~0.4250~~ *0.4295*	≤ 0.4250	≤ ~~0.4190~~ *0.4230*	≤ 0.4250	≤ 0.3840	≤ 0.3810
Pellet Dia. (in.)	≤ ~~0.4160~~ *0.4195*	≤ 0.4160	≤ ~~0.4110~~ *0.4140*	≤ 0.4160	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	~~0.636 - 0.641~~ ≤ *0.642*	~~0.636 - 0.641~~ ≤ *0.641*	0.640	0.640	0.566	~~0.569~~ ≤ *0.572*
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods	1	2	1 - 4 (note 5)	5	2	1 (note 4)
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	> 0.00	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.120	≤ 0.120

Notes:

1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
4. Square, replacing nine fuel rods.
5. Variable.
6. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array/Class	9x9 C	9x9 D	9x9 E	9x9 F	10x10 A
Clad Material (note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.)	≤ ~~173~~ *177*	≤ ~~170~~ *177*	≤ ~~170~~ *177*	≤ ~~170~~ *177*	≤ ~~182~~ *186*
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (note 3)
Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ ~~0.3590~~ *0.3640*	≤ ~~0.3810~~ *0.3860*	≤ 0.3520
Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ ~~0.3525~~ *0.3530*	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	0.572	0.572	0.572	0.572	0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	≥ ~~0.0305~~ *0.0300*	≥ ~~0.0305~~ *0.0120*	≥ ~~0.0305~~ *0.0120*	≥ 0.030
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ ~~0.100~~ *0.120*	≤ ~~0.100~~ *0.120*	≤ 0.120

Notes:
1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
4. Description of the fuel assembly class designation is provided in Chapter 6.

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

Fuel Assembly Array/Class	10x10 B	10x10 C	10x10 D	10x10 E	*8x8 F*
Clad Material (note 2)	Zr	Zr	SS	SS	*Zr*
Design Initial U (kg/assy.)	≤ ~~182~~ *186*	≤ ~~180~~ *186*	≤ 125	≤ 125	≤ *185*
Maximum Planar-Average Initial Enrichment (wt % ^{235}U)	≤ 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*
No. of Fuel Rods	91/83 (note 3)	96	100	96	*64*
Clad O.D. (in.)	≥ 0.3957	≥ ~~0.3790~~ *0.3780*	≥ 0.3960	≥ 0.3940	≥ *0.4576*
Clad I.D. (in.)	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500	≤ *0.3996*
Pellet Dia. (in.)	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430	≤ *0.3913*
Fuel Rod Pitch (in.)	0.510	0.488	0.565	0.557	≤ *0.609*
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 83	≤ 83	≤ *150*
No. of Water Rods	1 (Note 4)	5 (Note 5)	0	4	*N/A (Note 7)*
Water Rod Thickness (in.)	> 0.00	≥ ~~0.034~~ *0.031*	N/A	≥ 0.022	≥ *0.0315*
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080	≤ *0.055*

Notes:

1. Initial uranium weights and all dimensions are design nominal values. Actual uranium weights may be higher, within the manufacturer's tolerance. 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. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
4. Square, replacing nine fuel rods.
5. One diamond shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
6. Description of the fuel assembly class designation is provided in Chapter 6.
7. *This assembly is known as "QUAD+". It has four rectangular water cross segments dividing the assembly into four quadrants.*

Table 1.2.13
CHARACTERISTICS FOR DESIGN BASIS INTACT ZIRCALOY CLAD FUEL ASSEMBLIES

	MPC-68	MPC-24
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	5.85	8.54
Max. assembly length† (in.)	176.2	176.8
Max. assembly weight†† (lb.)	700	1680
Max. active fuel length† (in.)	150	150
Fuel rod clad material	zircaloy	zircaloy
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
	MPC-68	**MPC-24**
Max. initial enrichment (w/o ^{235}U)	Table 1.2.11	Table 1.2.10
Max. heat generation (W)	Figure 1.2.12 115 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) *183.5 (Assembly Class 8x8F)*	Figure 1.2.12
Min. cooling time (years)	Table 1.2.20 18 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) *10 (Assembly Class 8x8F)*	Table 1.2.20
Max. average burnup (MWD/MTU)	Table 1.2.20 30,000 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) *27,500 (Assembly Class 8x8F)*	Table 1.2.20
Min. initial enrichment (w/o ^{235}U)	Table 1.2.20 1.8 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A)	Table 1.2.20

† Unirradiated design dimensions are shown.

†† Fuel assembly weight including hardware based on DOE MPC DPS [1.1.1].

Table 1.2.16

SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length w/o C.C.† (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

† C.C. is an abbreviation for Control Components. *Fuel assemblies with control components may require shorter fuel spacers. Each user shall specify the fuel spacer lengths based on their fuel length and any control components and allowing an approximate 2 inch gap.*

Table 1.2.17

SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

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	23.6
Humboldt Bay	95	8	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	144.5†	11.2	110	17	14.5
Humboldt Bay Damaged Fuel or Fuel Debris	105.5†	8	79	35.25	35.25
LaCrosse	102.5	10.5	83	37	37.5

Note: *Each user shall specify the fuel spacer lengths based on their fuel length and allowing an approximate 2 inch gap.*

† Fuel length includes the damaged fuel container.

Table 1.2.19

DESIGN CHARACTERISTICS FOR STAINLESS STEEL CLAD FUEL ASSEMBLIES

	BWR MPC-68	PWR MPC-24
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	5.62	8.42
Max. assembly length† (in.)	102.5	138.8
Max. assembly weight†† (lb.)	400	1421
Max. active fuel length† (in.)	83	122
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
Max. heat generation (W)	83	488
Min. cooling time (yr)	~~15~~ *16*	~~16~~ *19* at 30,000 MWD/MTU ~~21~~ *24* at 40,000 MWD/MTU
Max. initial enrichment (wt.% ^{235}U)	4.0	4.0
Max. burnup (MWD/MTU)	22,500	40,000
Min. initial enrichment (wt.% ^{235}U)	3.5	3.1

† Dimensions are unirradiated nominal dimensions.

†† Fuel assembly weight including hardware based on DOE MPC DPS [1.1.1].

Table 1.2.21

DESIGN CHARACTERISTICS FOR THORIA RODS IN D1 THORIA ROD CANISTERS

PARAMETER	***MPC-68 or MPC-68F***
Cladding Type	*Zircaloy (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	*≤18*
Decay Heat Per Thoria Canister	*≤115 watts*
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	*Cooling time ≥18 years and average burnup ≤16,000 MWD/MTIHM*
Initial Heavy Metal Weight	*≤27 kg/canister*
Fuel Cladding O.D.	*≥0.412 inches*
Fuel Cladding I.D.	*≤0.362 inches*
Fuel Pellet O.D.	*≤0.358 inches*
Active Fuel Length	*≤111 inches*
Canister Weight	*≤550 lbs., including Thoria Rods*

1.3 DESIGN CODE APPLICABILITY

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997 [1.3.1], is the governing code for the structural design of the HI-STAR 100 System. The ASME Code is applied to each component consistent with the function of the component. Table 1.3.3 lists each structure, system and component (SSC) of the HI-STAR 100 System which are labeled Important to Safety (ITS), along with its function and governing Code. Some components perform multiple functions and in those cases, the most restrictive Code is applied. In accordance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components" [1.3.2] and according to importance to safety, components of the HI-STAR 100 System are classified as A, B, C, or NITS (not important to safety) in Table 1.3.3.

Table 1.3.1 lists the applicable ASME Code section and paragraph for material procurement, design, fabrication and inspection of the components of the HI-STAR 100 System that are governed by the ASME Code. The ASME Code section listed in the design column is the section used to define allowable stresses for structural analyses.

Table 1.3.2 lists the exceptions to the ASME Code for the HI-STAR 100 System and the justification for those exceptions.

The MPC is classified as important to safety. The MPC structural components include the internal fuel basket and the enclosure vessel. The fuel basket is designed and fabricated as a core support structure, in accordance with the applicable requirements of Section III, Subsection NG of the ASME Code, to the maximum extent practicable, as discussed in Table 1.3.2. The enclosure vessel is designed and fabricated as a Class 1 component pressure vessel in accordance with Section III, Subsection NB of the ASME Code, to the maximum extent practicable, as discussed in Table 1.3.2. The principal exceptions are the MPC lid, vent and drain cover plates, and closure ring welds to the MPC lid and shell, as discussed in Table 1.3.2. In addition, the threaded holes in the MPC lid are designed in accordance with the requirements of ANSI N14.6 [1.3.3] for critical lifts to facilitate vertical MPC transfer.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis, as presented in Chapter 2. The MPC closure ring welds are inspected by performing a liquid penetrant examination of the root pass *(if more than one weld pass is required)* and final weld surface, in accordance with the requirements contained in Section 7.5. The MPC lid weld may be examined by either volumetric or multi-layer liquid penetrant examination. If volumetric examination is used, it shall be the ultrasonic method and shall include a liquid penetrant examination of the root and final weld layers. If multi-layer liquid penetrant examination is used alone, at a minimum, it must include the root and final weld layers and each 3/8 inch of weld to detect critical weld flaws. The integrity of the MPC lid weld is further verified by performing a hydrostatic pressure test and a helium leak test in accordance with the requirements contained in Section 7.5.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, hydrostatic pressure testing, and helium leak testing performed during MPC fabrication

Table 1.3.2

LIST OF ASME CODE EXCEPTIONS FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Lid to Shell	*NB-5230*	*Radiographic (RT) or ultrasonic (UT) examination required*	*Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.*

Table 1.3.2 (continued)

LIST OF ASME CODE EXCEPTIONS FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root *(if more than one weld pass is required)* and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.
MPC Lid Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	If multi-layer liquid penetrant examination is used alone, at a minimum, it will include the root and final weld layers and each 3/8 inch of weld to detect critical weld flaws.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be hydrostatically tested as defined in Chapter 8. Accessibility for leakage inspections preclude a Code compliant hydrostatic test. All MPC vessel welds (except closure ring and vent/drain cover plate) are inspected by RT or UT. The vent/drain cover plate welds are confirmed by helium leakage testing and liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant.

TABLE 1.3.3

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

OVERPACK (1,2)

Primary Function	Component (3)	Safety Class (4)	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
			Subsection NF			surface with Carboline 890.	
Structural Integrity	Pocket Trunnion	B	ASME Section III; Subsection NF	SA705-630	Table 2.3.5	NA	NA
Structural Integrity	Lifting Trunnion	A	ANSI N14.6	SB637-N07718	Table 2.3.5	NA	NA
~~Structural Integrity~~	~~Backing Strip~~	~~B~~	~~Non-code~~	~~A569~~	~~Not required~~	~~NA~~	~~NA~~
~~Structural Integrity~~	~~Rupture Disk Coupling~~	~~NITS~~	~~Non-code~~	~~C/S~~	~~Not required~~	~~NA~~	~~NA~~
Structural Integrity	Rupture Disk	C	Non-code	~~Brass~~ *Commercial*	Not required	NA	Brass-C/S
~~Structural Integrity~~	~~Rupture Disk Pipe~~	~~C~~	~~Non-code~~	~~SA-106~~	~~Not required~~	~~NA~~	~~NA~~
Structural Integrity	Rupture Disk Plate	C	Non-code	A569	Not required	NA	NA
Structural Integrity	Removeable Shear Ring Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Structural Integrity	Thermal Expansion Foam	NITS	Non-code	Silicone Foam	Not required	NA	NA
Structural Integrity	Closure Bolt Washer	~~C~~ *NITS*	Non-code	~~SA240-304~~ *S/S*	Not required	NA	NA
Structural Integrity	Enclosure Shell Panels	B	ASME Section III; Subsection NF	SA515-70	Table 2.3.3	Paint outside surface with Carboline 890.	NA
Structural Integrity	Enclosure Shell Return	B	ASME Section III; Subsection NF	SA515-70	Table 2.3.3	Paint outside surface with Carboline 890.	NA
Structural Integrity	Port Cover	B	ASME Section III;	SA203E	Table 2.3.4	Paint outside	NA

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in NUREG/CR-6407. NITS stands for Not Important To Safety.

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

OVERPACK (1,2)

Primary Function	Component (3)	Safety Class (4)	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
			Subsection NF			surface with Carboline 890.	
Structural Integrity	Port Cover Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Operations	Trunnion Locking Pad and End Cap Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Operations	Lifting Trunnion End Cap	C	Non-code	SA516-70	Table 2.3.2	Paint exposed surfaces with Carboline 890.	NA
Operations	Lifting Trunnion Locking Pad	C	Non-code	SA516-70	Table 2.3.2	Paint exposed surfaces with Carboline 890.	NA
~~Operations~~	~~Alignment Pin~~	~~NITS~~	~~Non-code~~	~~SA193-B7~~	~~Not required~~	~~NA~~	~~NA~~
Operations	Nameplate	NITS	Non-code	SA240-304	Not required	NA	NA

Notes:
1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A,B and C denote important to safety classifications as described in NUREG/CR-6407. NITS stands for Not Important To Safety.

1.4 GENERAL ARRANGEMENT DRAWINGS and BILLS-OF-MATERIAL

The following drawings provide sufficient detail to describe the HI-STAR 100 packaging.

The classification of all components important to safety in accordance with Regulatory Guide 7.10 and NUREG/CR-6407 is provided in Table 1.3.3. Operational information, such as bolt torque and pressure-relief specifications are provided in Chapters 7 and 8. The maximum weight of the package and the maximum weight of the contents is provided in Table 1.2.1.

The following detailed HI-STAR 100 System design drawings are provided in this section:

Drawing Number/Sheet	Description	Rev.
5014-1395 Sht 1/4	HI-STAR 100 MPC-24 Construction	1~~0~~1
5014-1395 Sht 2/4	HI-STAR 100 MPC-24 Construction	~~9~~10
5014-1395 Sht 3/4	HI-STAR 100 MPC-24 Construction	~~9~~10
5014-1395 Sht 4/4	HI-STAR 100 MPC-24 Construction	~~8~~9
5014-1396 Sht 1/6	HI-STAR 100 MPC-24 Construction	1~~2~~3
5014-1396 Sht 2/6	HI-STAR 100 MPC-24 Construction	~~9~~10
5014-1396 Sht 3/6	HI-STAR 100 MPC-24 Construction	~~9~~10
5014-1396 Sht 4/6	HI-STAR 100 MPC-24 Construction	~~8~~
5014-1396 Sht 5/6	HI-STAR 100 MPC-24 Construction	8
5014-1396 Sht 6/6	HI-STAR 100 MPC-24 Construction	~~7~~8
5014-1397 Sht 1/7	Cross Sectional View of HI-STAR 100 Overpack	1~~4~~5
5014-1397 Sht 2/7	Detail of Top Flange & Bottom Plate of HI-STAR 100 Overpack	1~~0~~1
5014-1397 Sht 3/7	Detail of Bolt Hole & Bolt of HI-STAR 100 Overpack	1~~0~~1
5014-1397 Sht 4/7	Detail of Closure Plate Test Port and Name Plate Detail of HI-STAR 100 Overpack	1~~1~~2
5014-1397 Sht 5/7	Detail of Lifting Trunnion & Locking Pad of HI-STAR 100 Overpack	~~8~~9
5014-1397 Sht 6/7	Detail of Shear Ring and Closure Plate Bolt Installation of HI-STAR 100 Overpack	~~8~~9
5014-1397 Sht 7/7	Detail of Shear Ring of HI-STAR 100 Overpack	~~8~~9

Drawing Number/Sheet	Description	Rev.
5014-1398 Sht 1/3	HI-STAR 100 Overpack Orientation	1~~2~~*3*
5014-1398 Sht 2/3	Detail of Drain & Rupture Disk of HI-STAR 100 Overpack	~~9~~*10*
5014-1398 Sht 3/3	Detail of Vent & Port Plug of HI-STAR 100 Overpack	~~8~~*9*
5014-1399 Sht 1/3	Section "G" - "G" of HI-STAR 100 Overpack	1~~0~~*2*
5014-1399 Sht 2/3	Section "X"-"X" & View "Y" of HI-STAR 100 Overpack	~~8~~*9*
5014-1399 Sht 3/3	Detail of Trunnion Pocket Forging of HI-STAR 100 Overpack	~~9~~*10*
5014-1401 Sht 1/4	HI-STAR 100 MPC-68 Construction	1~~1~~*2*
5014-1401 Sht 2/4	HI-STAR 100 MPC-68 Construction	~~8~~*9*
5014-1401 Sht 3/4	HI-STAR 100 MPC-68 Construction	~~9~~*10*
5014-1401 Sht 4/4	HI-STAR 100 MPC-68 Construction	~~8~~*9*
5014-1402 Sht 1/6	HI-STAR 100 MPC-68 Construction	1~~3~~*4*
5014-1402 Sht 2/6	HI-STAR 100 MPC-68 Construction	1~~1~~*2*
5014-1402 Sht 3/6	HI-STAR 100 MPC-68 Construction	1~~1~~*2*
5014-1402 Sht 4/6	HI-STAR 100 MPC-68 Construction	~~9~~*10*
5014-1402 Sht 5/6	HI-STAR 100 MPC-68 Construction	9
5014-1402 Sht 6/6	HI-STAR 100 MPC-68 Construction	~~7~~*8*
5014-1763 Sht 1/1	HI-STAR 100 Assembly	~~3~~*4*
5014-1765 Sht 1/7	HI-STAR 100 Impact Limiter	1~~2~~*3*
5014-1765 Sht 2/7	HI-STAR 100 Bottom Impact Limiter	~~9~~*10*
5014-1765 Sht 3/7	HI-STAR 100 Top Impact Limiter	5
5014-1765 Sht 4/7	HI-STAR 100 Top Impact Limiter	1~~0~~*1*
5014-1765 Sht 5/7	HI-STAR 100 Top Impact Limiter Detail of Item #6	4
5014-1765 Sht 6/7	HI-STAR 100 Impact Limiter Honeycomb Details	~~1~~*2*
5014-1765 Sht 7/7	HI-STAR 100 Bottom Impact Limiter	4
5014-1782 Sht 1/1	HI-STAR 100 Assembly For Transport	~~1~~*2*

Drawing Number/Sheet	Description	Rev.
5014-1783 Sht 1/1	General Arrangement Damaged Fuel Container	~~1~~*3*
5014-1784 Sht 1/1	Damaged Fuel Container Details	~~0~~*2*
5014-1809 Sht 1/1	HI-STAR 100 Transportation Package Concept	1
BM-1476, Sht 1/2	Bills-of-Material for HI-STAR 100 Overpack	1~~2~~*3*
BM-1476, Sht 2/2	Bills-of-Material for HI-STAR 100 Overpack	1~~3~~*5*
BM-1478, Sht 1/2	Bills-of-Materials for 24-Assembly HI-STAR 100 PWR MPC	~~9~~*10*
BM-1478, Sht 2/2	Bills of Material for 24-Assembly HI-STAR 100 PWR MPC	1~~1~~*2*
BM-1479, Sht 1/2	Bills-of-Material for 68-Assembly HI-STAR 100 BWR MPC	1~~0~~*1*
BM-1479, Sht 2/2	Bills-of-Material for 68-Assembly HI-STAR 100 BWR MPC	1~~3~~*4*
BM-1819, Sht 1/1	Bills-of-Materials for HI-STAR 100 System Failed Fuel Canister	~~1~~*2*
9317.1-120-2	*D-1 Canister Assembly*	*0*
9317.1-120-3	*D-1 Canister Lid Assembly*	*1*
9317.1-120-4	*D-1 Canister Body*	*0*
9317.1-120-5	*D-1 Canister Bottom Assembly*	*1*
9317.1-120-6	*D-1 Canister Lower Lid Box Assy*	*1*
9317.1-120-7	*D-1 Canister Bumper Plate*	*0*
9317.1-120-8	*D-1 Canister Bale*	*0*
9317.1-120-9	*D-1 Canister Hanger*	*1*
9317.1-120-10	*D-1 Canister Lid Box*	*1*
9317.1-120-11	*D-1 Canister Lid Frame*	*1*
9317.1-120-13	*D-1 Canister Guide Bar*	*0*
9317.1-120-14	*D-1 Canister Fuel Support Plate*	*0*
9317.1-120-15	*D-1 Canister Screen Support Plate*	*0*
9317.1-120-17	*D-1 Canister Strut*	*0*

Drawing Number/Sheet	Description	Rev.
9317.1-120-18	*D-1 Canister Screen*	*1*
9317.1-120-19	*D-1 Canister Thin (Inner) Wiper*	*2*
9317.1-120-20	*D-1 Canister Spacer*	*0*
9317.1-120-21	*D-2/3 Fuel Spacer for TN-9 Cask*	*0*
9317.1-120-22	*D-1 Canister Thick Wiper*	*0*
9317.1-120-23	*D-1 Canister Lid Assembly For Failed Fuel*	*0*
9317.1-182-1	*Thoria Rod Canister Assembly*	*1*
9317.1-182-2	*Thoria Rod Canister Spacer*	*1*
9317.1-182-3	*Thoria Rod Canister Separator Plate Assembly*	*1*
9317.1-182-4	*Thoria Rod Canister Retaining Plate Assembly*	*1*
9317.1-182-5	*Thoria Rod Canister Retaining Plate Details*	*1*
9317.1-182-6	*Thoria Rod Canister Parts List*	*2*

Table 1.B.1

PROPERTIES OF HOLTITE-A NEUTRON SHIELD

PHYSICAL PROPERTIES (Reference: NAC International Brochure)	
% ATH	62 maximum (confirmed by Holtec in independent analyses)
Specific Gravity	1.68 g/cc ~~maximum~~ *nominal*
Thermal Conductivity	0.373 Btu/hr/ft-°F
Max. Continuous Operating Temperature	300°F
Specific Heat†	0.39 Btu/lb-°F
Hydrogen Density	0.096 g/cc minimum (confirmed by Holtec in independent analyses)
Radiation Resistance	Excellent
Ultimate Tensile Strength	4,250 psi
Tensile elongation	0.65%
Ultimate Compression Strength	10,500 psi
Compression Yield Strength	8,780 psi
Compression Modulus	561,000 psi
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5 (confirmed by Holtec)
wt% Hydrogen	6.0 (confirmed by Holtec)
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	up to 6.5 (Holtite-A uses 1% B_4C)

† BISCO Products Data from Docket M-55, NAC-STC TSAR.

APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA
(Total of 8 Pages Including This Page)

The information provided in this appendix specifies the thermal expansion foam (silicone sponge), paint, and anti-seize lubricant properties and demonstrates their suitability for use in spent nuclear fuel storage casks. The following is a listing of the information provided.

- HT-800 Series, Silicone Sponge, Bisco Products Technical Data Sheet
- Thermaline 450, Carboline, Product Data Sheet and Application Instructions
- Carboline 890, Carboline, Product Data Sheet and Application Instructions
- FEL-PRO Technical Bulletin, N-5000 Nickel Based-Nuclear Grade Anti-Seize Lubricant

HT-870 silicone sponge is specified as a thermal expansion foam to be placed in the overpack outer enclosure with the neutron shield. Due to differing thermal expansion of the neutron shield and outer enclosure carbon steel, the silicone sponge is provided to compress and allow the neutron shield material to expand. The compression-deflection physical properties are provided for the silicone sponge.

Silicone has a long and proven history in the nuclear industry. Silicone is highly resistant to degradation as a result of radiation at the levels required for the HI-STAR 100 System. Silicone is inherently inert and stable and will not react with the metal surfaces or neutron shield material. Additionally, typical operating temperatures for silicone sponges range from -50°F to 400°F.

Thermaline 450 is specified to coat the inner cavity of the overpack and Carboline 890 is specified to coat the external surfaces of the overpack. As can be seen from the product data sheets, the paints are suitable for the design temperatures (see Table 2.2.3) and chemical environment.

Nuclear grade anti-seize lubricant, N-5000, from FEL-PRO is specified as the lubricant for the overpack closure bolts. The lubricant is formulated to have the lowest practical levels of halogens, sulfur, and heavy metals. *NEVER-SEEZ NGBT provides equivalent properties to FEL-PRO N-5000 and is also acceptable for use on the HI-STAR 100 System.*

CHAPTER 2: STRUCTURAL EVALUATION

This chapter presents a synopsis of the evaluations carried out to establish the mechanical and structural characteristics of the HI-STAR 100 package as they pertain to demonstrating compliance with the provisions of 10CFR71. All required structural design analyses of the packaging, components, and systems Important to Safety (ITS) pursuant to the provisions of 10CFR71are documented in this chapter. The objectives of this chapter are twofold:

a. To demonstrate that the structural performance of the HI-STAR 100 package has been adequately evaluated for the conditions specified under normal conditions of transport and hypothetical accident conditions.

b. To demonstrate that the HI-STAR 100 package design has adequate structural integrity to meet the regulatory requirements of 10CFR71 [2.1.1].

To facilitate regulatory review, the assumptions and conservatism inherent in the analyses are identified along with a complete description of the analytical methods, models, and acceptance criteria. A summary of other considerations germane to satisfactory structural performance, such as corrosion and material fracture toughness is also provided.

Detailed numerical computations supporting the conclusions in the main body of this chapter are further supplemented through a series of appendices. Where appropriate, the text within the chapter makes reference to the information contained in the appendices. Section 2.10 contains the complete list of appendices that support this chapter.

This SAR is written to conform to the requirements of NUREG-1617 and 10CFR71 and follows the format of Regulatory Guide 7.9 [1.0.3]. It is noted that the areas of NRC staff technical inquiries with respect to 10CFR71 structural compliance span a wide array of technical topics within and beyond the material in this chapter. To facilitate the staff's review, Table 2.0.1 "Matrix of NUREG-1617/10CFR71 Compliance - Structural Review", is included in this chapter. A comprehensive cross-reference of the topical areas set forth in Section 2.3.2 (Regulatory Requirements) of the draft Regulatory Guide 1617, along with the sponsoring paragraphs in10CFR71, and the location of the required compliance information, within this SAR, is contained in Table 2.0.1.

Section 2.10.2 contains a summary of the evaluation findings derived from the technical information presented in this chapter.

TABLE 2.0.1- MATRIX OF NUREG-1617/10CFR71 COMPLIANCE – STRUCTURAL REVIEW †

SECTION IN NUREG-1617 AND APPLICABLE 10CFR71/REG.GUIDE (R.G.) SECTIONS	NUREG-1617/10CFR71 COMPLIANCE ITEM	LOCATION IN SAR CHAPTER 2	LOCATION OUTSIDE OF SAR CHAPTER
2.3.1 Description of Structural Design			
10CFR71.31(a)(1); 10CFR71.33	Description of Structural Design	2.1	1.2.3
10CFR71.33	Drawings		1.4
10CFR71.33	Weights and Center of Gravity	2.2	
10CFR71.31(c)	Applicable Codes/Standards		1.3
2.3.2 Material Properties			
10CFR71.33	Materials and Material Specifications	2.3	
10CFR71.33	Prevention of Chemical, Galvanic, or Other Reactions	2.4	
10CFR71.43(d)	Effects of Radiation on Materials	2.4.4	

TABLE 2.0.1- MATRIX OF NUREG-1617/10CFR71 COMPLIANCE – STRUCTURAL REVIEW (Continued)

SECTION IN NUREG-1617 AND APPLICABLE 10CFR71/REG.GUIDE (R.G.) SECTIONS	NUREG-1617/10CFR71 COMPLIANCE ITEM	LOCATION IN SAR CHAPTER 2	LOCATION OUTSIDE OF SAR CHAPTER
R.G 7.11, 7.12	Brittle Fracture	2.1.2.3	
2.3.3 Lifting and Tie Down Standards for All Packages			
10CFR71.45(a)	Lifting Devices	2.5; 2.A; 2.B; 2.S; 2.T; 2.AG	1.4
10CFR71.45(b)	Tie-Down Devices	2.5; 2.C; 2.R	1.4
2.3.4 General Considerations for Structural Evaluation of Packaging			
10CFR71, Subpart E,F	Evaluation by Analysis		
10CFR71.35(a), 71.41(a)	• Models, Methods, and Results	2.6, 2.7.1,2.72	
10CFR71, Subpart E,F	• Material Properties	2.3	
"	• Boundary Conditions	2.6	
"	• Dynamic Amplifiers	2.K	
"	• Load Combinations	2.1	
"	• Margins of Safety	2.5, 2.6, 2.7	

TABLE 2.0.1- MATRIX OF NUREG-1617/10CFR71 COMPLIANCE – STRUCTURAL REVIEW (Continued)

SECTION IN NUREG-1617 AND APPLICABLE 10CFR71/REG.GUIDE (R.G.) SECTIONS	NUREG-1617/10CFR71 COMPLIANCE ITEM	LOCATION IN SAR CHAPTER 2	LOCATION OUTSIDE OF SAR CHAPTER
10CFR71, Subparts E,F	Evaluation by Test		
10CFR71.73(a)	• Procedures for Impact Testing	2.7.1; 2.H	
"	• Test Specimens	2.7.1; 2.H	
10CFR71.73(c)(1)	• Drop Orientations	2.7.1; 2.H	
"	• Conclusions	2.7.1; 2.H	
2.3.5 Normal Conditions of Transport			
10CFR71.71 with reference to 10CFR71 sections 71.35(a), 71.43(f), 71.51(a)(1), 71.55(d)(4)	Heat	2.6.1; 2.D; 2.E; 2.F; 2.J; 2.L; 2.N; 2.O;2.Q; 2.U;2 AC; 2.AE	
"	Cold	2.6.2; 2.AE; 2.AI; 2.AJ; 2.AK	
"	Reduced External Pressure	2.6.3	
"	Increased External Pressure	2.6.4	
"	Vibration	2.6.5	
"	Water Spray	2.6.6	
"	Free Drop	2.6.1; 2.6.2; 2.6.7; 2.AE	
"	Corner Drop	NA	NA
"	Compression	NA	NA
"	Penetration	NA	NA

TABLE 2.0.1- MATRIX OF 10CFR71 COMPLIANCE – STRUCTURAL REVIEW (Continued)

SECTION IN NUREG-1617 AND APPLICABLE 10CFR71/REG.GUIDE (R.G.) SECTIONS	NUREG-1617/10CFR71 COMPLIANCE ITEM	LOCATION IN SAR CHAPTER 2	LOCATION OUTSIDE OF SAR CHAPTER
2.3.6 Hypothetical Accident Conditions			
10CFR71.73(c)(1)	Free Drop	2.7.1; 2.I; 2.J; 2.L; 2.N; 2.O; 2.U; 2.AC; 2.AE; 2.AF; 2.AH; 2.AO	
10CFR71.73(c)(2)	Crush	NA	NA
10CFR71.73(c)(3)	Puncture	2.7.2; 2.U	
10CFR71.73(c)(4)	Thermal	2.7.3;2.G; 2.J; 2.L; 2.N	
10CFR71.73(c)(5)	Immersion-Fissile Material	2.7.4	NA
10CFR71.73(c)(6)	Immersion – All Material	2.7.5; 2.J	
2.3.7 Special Requirements for Irradiated Nuclear Fuel Shipments			
10CFR71.61	Elastic Stability of Containment	2.7.5; 2.J	
"	Closure Seal Region Below Yield Stress	2.AE	
2.3.8 Internal Pressure Test			
10CFR71.85(b)	Internal Pressure Test – All stresses below yield	2.6.1.4.3	8.1

TABLE 2.0.1- MATRIX OF 10CFR71 COMPLIANCE – STRUCTURAL REVIEW (Continued)

SECTION IN 10CFR71	10CFR71 COMPLIANCE ITEM	LOCATION IN SAR CHAPTER 2	LOCATION OUTSIDE OF SAR CHAPTER
Appendices			
	Supplemental Information	2.10	

† Legend for Table 2.0.1

Per the nomenclature defined in Chapter 1, the first digit refers to the chapter number, the second digit is the section number within the chapter; an alphabetic character in the second place means it is an appendix to the chapter.

NA Not Applicable for this item

Table 2.2.3

CALCULATED MAXIMUM LIFT WEIGHT ON CRANE HOOK ABOVE POOL

Item	Weight (lb)
Total weight of overpack	153,710
Total weight of MPC(upper bound) + fuel	89,057[1]
Overpack closure plate	-7,984
Water in MPC and overpack	16,384
Lift yoke	3,600
Inflatable annulus seal	50
TOTAL	254,817[2]

1 Includes MPC closure ring.

2 Trunnions are rated to lift 250,000 lbs. For weight exceeding 250,000lbs, weight can be reduced by partial draining of the MPC or by limiting the number of control components. See Chapter 7 for operational controls.

Table 2.2.4

COMPONENT WEIGHTS AND DIMENSIONS FOR ANALYTIC CALCULATIONS*

Component	Weight (lbs)
MPC baseplate	3,000
MPC closure lid	10,400
MPC shell	5,900
MPC miscellaneous parts	3,700
Fuel basket	13,000
Fuel	54,000
Total MPC	90,000
Overpack baseplate	10,000
Overpack closure plate	8,000
Overpack shell	137,000
Total overpack	155,000
Total HI-STAR 100 lift weight	250,000
Impact limiters	37,000
HI-STAR with limiters	282,000
Item	**Dimension (inch)**
Overpack Outer Diameter	96
Overpack Length	203.125
MPC Outer Diameter	68.375
MPC Length	190.5
Overpack Inner Diameter	68.75

* Note: Analytical calculations may use weights and dimensions in Table 2.2.4 or actual weights and dimensions for conservatism in calculation of safety factors. Finite element analyses may use weights calculated based on input weight densities.

assume in the calculation that the rotation trunnions carry 100% of the HI-STAR 100 weight during rotation, which is further magnified by a dynamic load factor of 1.15. The results from the weld analyses in Appendix 2.R are summarized in the table below:

Rotation Trunnion Weld Group Safety Factors per 10CFR71 Requirements			
Item	Required Weld Size (inch)	Available Weld Size (inch)	Safety Factor
Transport Tie-Down - 10CFR71.45(b)	3.19	4.25	1.33
Transport Rotation - 10CFR71.45(a)	0.831	4.25	5.11

Safety factors are greater than 1.0 indicating that the requirements of 10CFR71.45 are met.

2.5.2.7.6 Structural Integrity of the Top Flange Shear Ring

Longitudinal transport loads on the HI-STAR 100 System that are directed towards the forward (top end of the package) are resisted by a shear ring, integral with the top flange, which extends around the top flange over a 140 degree arc. Figure 2.5.6 shows the shear ring. The following dimensions are applicable:

$R = 41.625$ in.
$t = 1.5$ in.
$L = 5.0$ in.

The shear ring resists the 10g longitudinal load from normal transport, designated here as, F_{long} = 2,820,000 lb., by developing a bearing pressure on the contact surface of the shear ring and then transferring the load through the available shear area. In the following, we determine the bearing and shear stresses that develop when the longitudinal load from normal transport is imposed. Figure 2.5.7 is a sketch showing the developed shear and bearing loads on the shear ring.

* Bearing stress on contact surface

The contact surface area is

2.10 MISCELLANEOUS ITEMS

2.10.1 Appendices†

This section contains a summary listing of all appendices to Chapter 2 of the SAR.

APPENDIX 2.A: TOP FLANGE BOLT HOLE ANALYSIS

APPENDIX 2.B: LIFTING TRUNNION STRESS ANALYSIS

APPENDIX 2.C: CALCULATION OF TRANSPORT TIE-DOWN REACTIONS

APPENDIX 2.D: HI-STAR 100 COMPONENT THERMAL EXPANSIONS; MPC-24

APPENDIX 2.E: DELETED

APPENDIX 2.F: HI-STAR 100 COMPONENT THERMAL EXPANSIONS; MPC-68

APPENDIX 2.G: THERMAL EXPANSION DURING FIRE ACCIDENT

APPENDIX 2.H: IMPACT LIMITER CHARACTERISTICS, DYNAMIC SIMULATION OF HYPOTHETICAL ACCIDENT EVENT, AND SCALE MODEL TESTS

APPENDIX 2.I: OVERPACK PROTECTION LIP DEFORMATION ANALYSIS

APPENDIX 2.J: CODE CASE N-284 STABILITY CALCULATIONS

APPENDIX 2.K: CALCULATION OF DYNAMIC FACTORS

APPENDIX 2.L: ANALYSIS OF MPC TOP CLOSURE

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† Some of the appendices have been created using the electronic scratchpad program MATHCAD that incorporates text and "live" calculations. Special symbols are used by the program for certain mathematical operations. One such symbol that appears in some of the appendices is a small solid block at the end of an equation. This symbol means that the equation is not evaluated at that location (i.e., is not "live" but is simply employed as "text".

2.10.2 Summary of NUREG -1617/10CFR71 Compliance

This subsection provides a "road map" of technical information to demonstrate that the SAR in compliance with the provisions of NUREG-1617 and associated referenced sections of 10CFR71 necessary to certify the HI-STAR 100 package for transport.

Description of Structural Design

The package structural design description and the contents of the application meet the requirements of 10CFR 71.31 and Regulatory Guide 7.9. Applicable sections where this is demonstrated are 1.2.1; 1.3; 1.4; and 2.1.

The codes and standards used in the package design are listed in 1.3. The use of the ASME

Boiler and Pressure Vessel Code is in compliance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage Components".

Material Properties

There are no significant chemical, galvanic or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry or wet environment conditions. The applicable subsection where this is demonstrated is 2.4.4.

The effects of radiation on materials are considered and package containment is constructed from materials that meet the requirements of Reg. Guides 7.11 and 7.12. Applicable subsections where this is demonstrated are: 1.2.1; 2.1.2; and, 2.4.4.

Lifting and Tie-Down Standards for All Packages

Lifting and Tie-Down systems meet 10CFR 71.45 standards. The applicable section where this is demonstrated is 2.5.

General Considerations for Structural Evaluation of Packaging

The packaging structural evaluation meets the requirements of 10CFR 71.35. Applicable chapters and/or sections where this is demonstrated are: 2.5; 2.6; 2.7

Normal Conditions of Transport

The packaging structural performance under normal conditions of transport demonstrate that there will be no substantial reduction in the effectiveness of the packaging. The applicable section where this is demonstrated is 2.6.

Hypothetical Accident Conditions

The packaging structural performance under the hypothetical accident conditions demonstrates that the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of 10CFR Part 71. The applicable section where this is demonstrated is 2.7.

Special Requirement for Irradiated Nuclear Fuel Shipments

The containment structure meets the 10CFR 71.61 requirements for irradiated nuclear fuel shipments. The applicable section where this is demonstrated is 2.7.

Internal Pressure Test

The containment structure meets the 10CFR 71.85(b) requirements for pressure test without yielding. The applicable subsection where this is demonstrated is 2.6.1.4.3.

APPENDIX 2.L: ANALYSIS OF MPC TOP CLOSURE

2.L.1 Scope

This appendix provides the stress analysis of the MPC top closure plate under bounding load cases for both storage and transport scenarios.

2.L.2 Methodology

Conservative values for stresses on the closure plate are obtained by using classical strength of materials formulations, which are sufficient for determining primary stresses in the component. The peripheral weld to the MPC shell is protected by a thin closure ring. The analysis of this ring is performed using a finite element model.

2.L.3 References

[2.L.1] S.P. Timoshenko, Strength of Materials, Vol. 2, Third Edition, Van Nostrand, 1956.

[2.L.2] ANSYS Finite Element Code, 5.3, Ansys, Inc., 1997.

2.L.4 Configuration, Geometry, and Input Weight Data

2.L.4.1 Configuration and Geometry

Figure 2.L.1 shows a sketch of the top closure lid with the the closure ring attached. The configuration is the same for all MPC types. The following dimensions are obtained from drawing no. 1393.

The outer radius of the lid, $R_{lid} := \frac{67.375}{2} \cdot in$

The inner radius of the closure ring, $R_i := \frac{53.03125}{2} \cdot in$

The outer radius of the closure ring, $R_o := \frac{67.875}{2} \cdot in$

The minimum thickness of the lid, $h := 9.5 \cdot in$

The closure ring thickness, $t := 0.375 \cdot in$

2.L.4.2 Input Weight Data

The bounding weight of the closure lid (MPC-68), $W_{lid} := 10400 \cdot lbf$ Table 3.2.4

The bounding weight per square inch of lid, $P_{lid} := \dfrac{W_{lid}}{\pi \cdot R_{lid}^2}$ $P_{lid} = 2.917 \cdot psi$

The bounding weight of the fuel basket plus fuel,

$W_{fuel} := 13000 \cdot lbf + 54000 \cdot lbf$ Table 3.2.4

The maximum total package weight of the MPC (including dynamic load factor),

$W_{lift} := 1.15 \cdot 90000 \cdot lbf$ Table 3.2.4

The maximum lifted weight is the bounding MPC weight with an applied 0.15 inertia load factor to bound loads during an MPC transfer operation.

2.L.5 Acceptance Criteria

Level A or Level D primary stress intensity levels must not be exceeded under the defined load conditions. Load cases considered are set to bound all requirements for either storage or transport.

2.L.6 Allowable Strengths

Allowable strengths at the design temperature of 550°F and at the accident temperature of 775°F are used. The material used is Alloy X. The relevant allowable stress intensities for primary membrane stress and for combined primary bending and primary membrane stress, for ASME Section III, Subsection NB components, are therefore:

The Level A allowable stress intensity for combined stress (550°F), $S_{ac} := 25450 \cdot psi$

The Level A allowable stress intensity for membrane stress (550°F), $S_{am} := 16950 \cdot psi$

The Level D allowable stress intensity for combined stress (550°F), $S_{dc} := 61050 \cdot psi$

The Level D allowable stress intensity for membrane stress (550°F), $S_{dm} := 40700 \cdot psi$

The Level D allowable stress intensity for combined (775°F), $S_{firec} := 54225 \cdot psi$

The Level D allowable stress intensity for membrane 775°F), $S_{firem} := 36150 \cdot psi$

The closure ring, which functions as the secondary seal for the MPC, is located on the upper surface of the lid. The appropriate design temperature at this location is 400°F, which bounds all non-accident metal temperatures obtained at that location in the analyses of Chapter 3. The Level A membrane and membrane plus bending allowable stress intensities at this temperature are:

$$S_{amr} := 18700 \cdot psi$$

$$S_{acr} := 28100 \cdot psi$$

2.L.7 Load Cases

The following bounding loads are considered as potential limiting loads for the top closure plate structural qualification. Only the most limiting combinations are used for the qualification. For calculation purposes, the applied loads are considered as equivalent surface pressures.

The external pressure, $P_{ext} := 125 \cdot psi$

The internal pressure, $P_{int} := 100 \cdot psi$

The fire pressure, $P_{fire} := 125 \cdot psi$

A bottom end drop on the overpack baseplate gives a pressure of,

$$P_{sd} := \frac{60 \cdot W_{lid}}{\pi \cdot {R_{lid}}^2} \qquad P_{sd} = 175.024 \cdot psi$$

A top end drop on the overpack closure plate gives a pressure,

$$P_{td} := \frac{60 \cdot W_{fuel}}{\pi \cdot {R_{lid}}^2} \qquad P_{td} = 1.128 \cdot 10^3 \cdot psi$$

The center lift weight, $P_{lift} := W_{lift}$

Note that external pressure never governs because internal pressure adds a membrane stress component. The center lift weight load is included to incorporate a fully-loaded lifting operation that is defined in the HI-STAR100 Storage TSAR (Docket 72-1008).

For the qualification of the closure ring, only a single load case need be considered. If the primary, load carrying MPC cover plate-to-MPC shell peripheral weld leaks, then the closure ring will be subjected to the internal pressure load, and behaves as an annular plate supported at its inner and outer periphery. While this case is amenable to manual calculations, the case is analyzed using the finite element method for simplicity.

2.L.8 Calculations

The stress analysis of the closure plate is performed by conservatively assuming that the closure plate acts as a simply supported plate. This will conservatively predict a higher stress at the center of the plate. In the plate analysis, it is assumed that the thickness is constant. This is slightly nonconservative at the outer periphery of the plate since the closure ring is a separate component; however, as will be seen from the results, the safety factors are large so that the effect is negligible.

In all of the following analyses, since the circumferential stress has the same sign as the radial stress, stress intensities differ from stresses only by the surface pressure, where applicable.

2.L.8.1 Level A Bounding Calculations

The design load is the internal pressure case, since there is a direct stress as well as a bending stress because of the peripheral weld. However, for a transfer operation, there exists the potential for a bounding Level A condition to be internal pressure plus a central lifted load.

2.L.8.1.1 Load Case E1.a, Table 2.1.7

This load case consists of internal pressure only. Reference [2.L.1] provides a formula for the maximum bending stress at the center of a simply supported circular plate. For the case of internal pressure alone, the stress intensity SI_1 and resultant safety factor are determined as: The Poisson's ratio of the material, $\nu := 0.3$

The bending stress due to internal pressure, $\sigma_b := \frac{3\cdot(3+\nu)}{8}\cdot(P_{int} + P_{lid})\cdot\left(\frac{R_{lid}}{h}\right)^2$

$$\sigma_b = 1601 \cdot psi$$

The direct stress due to internal pressure, $\sigma_d := -P_{int}$ $\qquad \sigma_d = -100 \cdot psi$

The combined stress intensity, $SI_1 := (\sigma_b + |\sigma_d|)$ $SI_1 = 1701 \cdot psi$

The factor of safety, $FS_1 := \frac{S_{ac}}{SI_1}$ $FS_1 = 15.0$

2.L.8.2 Level D Bounding Calculations

2.L.8.2.1 Load Case E3.a, Table 2.1.7

2.L.8.2.1.1 Bounding 10CFR72 (Storage) Bottom End Drop

This load case corresponds to the 10CFR72 (storage) end drop on the overpack baseplate. The amplified weight of the lid, plus the external design pressure, give rise to a bending stress. This bending stress and the resultant safety factor are determined as:

The bending stress due to the loading, $\sigma_b := \frac{3 \cdot (3 + \nu)}{8} \cdot (P_{sd} + P_{ext}) \cdot \left(\frac{R_{lid}}{h}\right)^2$

$\sigma_b = 4669 \cdot psi$

The factor of safety, $FS_3 := \frac{S_{dc}}{\sigma_b}$ $FS_3 = 13.1$

2.L.8.2.1.2 Bounding 10CFR71 (Transport) Top End Drop

For this case, the MPC closure plate is supported by the overpack closure plate over a peripheral band of support. It is conservative for the MPC qualification to assume that all support is at the outer edge. Therefore, the bending stress and resultantsafety factor due to the equivalent pressure of the fuel basket and fuel, the applied weight of the closure plate and the internal pressure is determined as:

The bending stress due to the loading, $\sigma_b := \frac{3 \cdot (3 + \nu)}{8} \cdot (P_{int} + P_{sd} + P_{td}) \cdot \left(\frac{R_{lid}}{h}\right)^2$

$\sigma_b = 21825 \cdot psi$

The factor of safety, $FS_4 := \frac{S_{dc}}{\sigma_b + P_{int}}$ $FS_4 = 2.8$

2.L.8.2.1.3 <u>Load Case E5, Table 2.1.7</u>

This load case considers dead load, fire pressure, and fire temperature material properties.

The bending stress is, $\sigma_b := \frac{3\cdot(3+\nu)}{8}\cdot\left(P_{fire} + P_{lid}\right)\cdot\left(\frac{R_{lid}}{h}\right)^2$

$\sigma_b = 1.991\cdot10^3\cdot psi$

The factor of safety is, $FS_5 := \frac{S_{firec}}{\sigma_b}$ $FS_5 = 27.2$

2.L.8.3 <u>Peripheral Weld Stress</u>

The area of the weld is computed by multiplying the total length of the weld (at radius R_{lid}) by the weld thickness. The weld capacity is found by multiplying this area by a quality factor (defined in ASME Subsection NG) and by the appropriate weld stress allowable from ASME Subsection NF. The weld between the MPC lid and the shell is a 3/4 inch J-groove weld. For conservatism, a smaller weld size (i.e., 5/8 inch) is considered in the following stress evaluations.

The thickness of the weld, $t_{weld} := 0.625\cdot in$

The quality factor for a single groove weld that is examined by root and final PT is $n := 0.45$

The allowable weld stresses for Level A and Level D conditions are Sa and Sd, respectively. The weld metal strength is assumed to decrease with temperature in the same manner as does the base metal (Alloy X)

$$Sa := 0.3\cdot70000\cdot\left[1 - \left(\frac{75 - 63.3}{75}\right)\right]\cdot psi \qquad Sa = 1.772\cdot10^4\cdot psi$$

$$Sd := .42 \cdot 70000 \cdot \left[1 - \left(\frac{75 - 63.3}{75}\right)\right] \cdot psi \qquad Sd = 2.481 \cdot 10^4 \cdot psi$$

The maximum load capacity of the weld, $LC_{weld} := n \cdot 2 \cdot \pi \cdot R_{lid} \cdot t_{weld} \cdot Sa$

$$LC_{weld} = 1.055 \cdot 10^6 \cdot lbf$$

The factor of safety of this load capacity, for the Level A center lift loading case (Load Case E2, Table 3.1.4 in Docket 72-1008), is determined as:

$$FS_6 := \frac{LC_{weld}}{W_{lift} + \pi \cdot P_{int} \cdot R_{lid}^2} \qquad FS_6 = 2.29$$

The bounding weld load for Level D conditions is determined by multiplying the equivalent pressure load for the load case by the area of the closure plate. The bottom end drop is taken by the welds, and the top end drop is taken by bearing on the overpack closure plate.

$$L_{weld} := P_{sd} \cdot \pi \cdot \left(R_{lid}\right)^2 \qquad L_{weld} = 624000 \cdot lbf$$

$$MS_7 := \frac{Sd}{Sa} \cdot \frac{LC_{weld}}{L_{weld}} - 1 \qquad MS_7 = 1.37$$

To further demonstrate the adequacy of the weld, its capacity is compared to a weld load that equals three times the total lifted weight. The factor of safety is

$$FS_8 := \frac{LC_{weld}}{3 \cdot W_{lift}} \qquad FS_8 = 3.40$$

2.L.8.4 Fatigue Analysis of Weld

The welds will be subjected to cyclic stress each time the MPC is lifted. The force difference is equal to W_{lift}. Pressure loads are not a fatigue consideration since they remain relatively constant during normal operation. Therefore, the effective fatigue stress can be determined as follows

The fatigue factor for a single groove weld that is examined by root and final PT is $f := 4$ and the alternating stress is

$$\sigma := \frac{\left(f \cdot \frac{W_{lift}}{2}\right)}{2 \cdot \pi \cdot R_{lid} \cdot t_{weld}} \qquad \sigma = 1565 \cdot psi$$

This stress is compared to curve B in Figure I-9.2.2 of the ASME Division I Appendices per Subsection NG. This curve shows that the welds have unlimited life at this stress level.

2.L.8.5 Closure Ring Analysis

The closure ring must be capable of withstanding the application of the full MPC internal pressure, to ensure that a leak in the primary closure plate weld will be contained. This condition is modeled as an annular ring subject to the design internal pressure. A finite element analysis of a thin ring with an applied pressure is performed using the ANSYS finite element code. The thin ring is simulated by four layers of PLANE42 axisymmetric quadrilateral elements (see Figure 2.L.2). The boundary condition is conservatively set as zero displacement at node locations 1 and 2 (see Figure 2.L.2). The bottom surface is subjected to a 100 psi pressure to simulate leakage of the primary MPC weld. The maximum stress intensity in the ring (occurring at the top center point) and the resultant factor of safety for Level A conditions are determined as:

The maximum stress intensity in the ring, $SI_{ring} := 20001 \cdot psi$

The factor of safety, $FS_9 := \frac{S_{acr}}{SI_{ring}}$ $\qquad FS_9 = 1.405$

Since the actual support condition provides a edge fixity condition that is intermediate between a simple support and a built-in support, this result is very conservative.

The total load capacity of the closure ring weld is determined by calculating the total area of the two weld lines at radii R_i and R_o, multiplying by the allowable weld stress, and conservatively applying the specified weld efficiency.

The closure ring weld thickness, $t_{crw} := 0.125 \cdot in$ (this allows for fit-up and also provides improved ALARA considerations)

The quality factor for a single groove or a single fillet weld that is examined by PT is taken as the same value used for the MPC Top Closure-to-shell weld: $n := 0.45$

The load capacity of the ring welds is,

$$LC_{crw} := n \cdot 2 \cdot \pi \cdot \left(R_i + \frac{R_o}{\sqrt{2}} \right) \cdot t_{crw} \cdot Sa \qquad LC_{crw} = 3.164 \cdot 10^5 \cdot lbf$$

The margin of safety of these welds for the applied loading condition (design internal pressure only) is determined as:

$$MS_{10} := \frac{LC_{crw}}{\pi \cdot P_{int} \cdot \left(R_o^{\,2} - R_i^{\,2} \right)} - 1 \qquad MS_{10} = 1.24$$

2.L.9 Conclusions

The results of the evaluations presented in this appendix demonstrate the adequacy of the MPC closure plate, closure ring and associated weldments to maintain their structural integrity during applied bounding load cases considered. Positive safety margins exist for all components examined for all load cases considered.

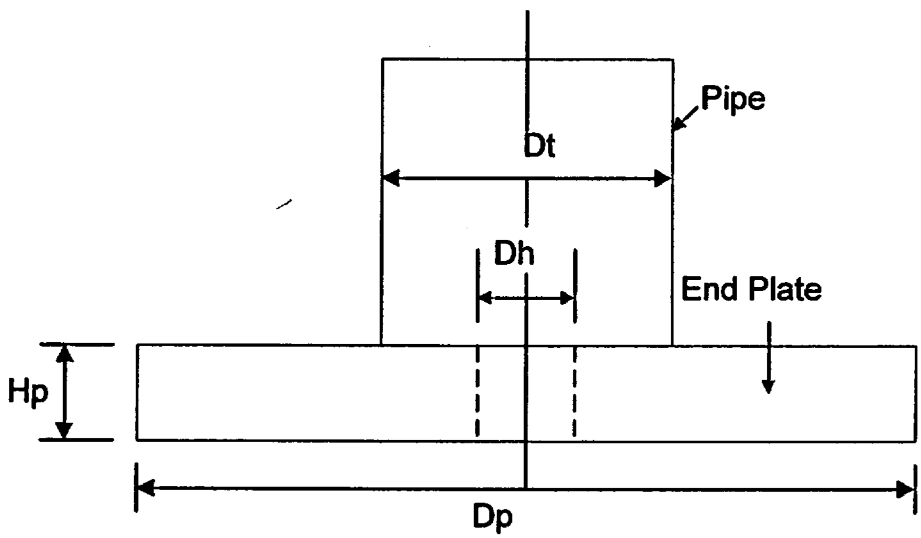
The bending stress evaluation of the closure ring conservatively assumes a simple support condition exists at the peripheral welds. The associted stress at the weld line is a radially symmetric primary shear stress that is well withn Code limits if a 0.125" (min.) weld is utilized.

The closure ring has been analyzed as a continuous ring with no discontinuities around the periphery. Practical considerations in fabrication mandate that the closure ring should be constructed from two or more sections connected by a radially oriented weld. In addition, to positively preclude any interaction with the peripheral weld in the MPC lid, the radial welds connecting radial segments of the closure ring to adjacent segments should be partial penetration welds. These radial welds are not required for satisfaction of equilibrium and generate no primary state of stress in the closure ring due to their presence. Small stress adders that develop due to the local structural discontinuity at the partial penetration weld locations produce peak stresses that have no prescribed limit in the ASME Code.

Indeed, the radial weld lines are in the category of "seal" welds that are essentially required to prevent leakage but bear little stress due to lateral pressure.

2.O.4 Analysis of Upper Spacer End Plate for PWR Spacers

Some PWR fuel types are not supportable by the current upper spacer design having a simple pipe extension. To insure that all PWR fuel types are captured, an end plate having sufficient diameter is welded to the end of the pipe to extend the contact area. This section of the appendix addresses the stress analysis of the end plate to insure that it performs as desired under a handling accident that results in a direct impact of the fuel assembly onto the end plate. The configuration is shown below:



The dimensions are:(note that outer adius is taken equal to inside radius of limiting fuel assembly contact circle

$Hp := 0.75 \cdot in$ $\quad Dp := 4.1 \cdot in$ $\quad Dt := 3.5 \cdot in$ $\quad Dh := 1 \cdot in$

Under the postulated handling accident, the total applied load is (design basis deceleration of 60 g's):

$P := 60 \cdot 1680 \cdot lbf \qquad P = 1.008 \times 10^5\, lbf$

This load may be applied as a line load around the outer periphery

$$q_o := \frac{P}{\pi \cdot Dp} \qquad q_o = 7.826 \times 10^3 \frac{lbf}{in}$$

or it may be applied as a line load at a diameter of 1.8" (from a survey of fuel assembly types)

$$q_i := \frac{P}{\pi \cdot 1.8 \cdot in} \qquad q_i = 1.783 \times 10^4 \frac{lbf}{in}$$

In either case, the shear load at the pipe connection is approximately

$$q_p := \frac{P}{\pi \cdot Dt} \qquad q_p = 9.167 \times 10^3 \frac{lbf}{in}$$

At the design temperature, the ultimate strength is, (conservatively neglect any increase in ultimate strength due to strain rate effects

$$S_u := 62350 \cdot psi$$

The spacer pipe has been designed to NG, Level D requirements for axial strength and to the appropriate ASME Code requirements for gross stability. The function of the end plate is to insure that the fuel assembly impacts the spacer; the only requirement is that under an accident condition, no permanent deformation of this end plate occurs to the extent that the positioning limits of the fuel assembly is compromised. This is insured if we demonstrate that the ultimate shear capacity of the added end plate and the ultimate moment capacity of the end plate is not exceeded during the impact. Satisfaction of these stress limits will insure that no large axial movement of the assembly can occur because of the impact.

The ultimate shear capacity of the section is taken as 0.577Su, and the ultimate moment capacity is calculated assuming perfectly plastic behavior at the ultimate stress. Therefore, at any section of the plate the shear capacity is:

$$q_{cap} := .577 \cdot S_u \cdot Hp \qquad q_{cap} = 2.698 \times 10^4 \frac{lbf}{in}$$

Comparison of this limit with the peripheral shear loads computed previously demonstrates that the end plate will not experience a gross shear failure at any section. The minimum safety factor "SF" is

$$\frac{q_{cap}}{q_i} = 1.514$$

The ultimate moment capacity is (assume rectangular distribution throuh the thickness):

$$M_u := S_u \cdot \frac{Hp^2}{4} \qquad M_u = 8.768 \times 10^3 \, in \cdot \frac{lbf}{in}$$

The weight of the added end plate is:

$$Weight := 0.29 \cdot \frac{lbf}{in^3} \cdot \frac{\pi}{4} \cdot Hp \cdot \left(Dp^2 - Dh^2\right) \qquad Weight = 2.701\,lbf$$

The following calculations are performed to establish the maximum bending moment in the end plate based on the two extreme locations of impact load. The electronic version of Roark's Handbook (6th Edition) that is a Mathcad add-on, is used for this computation. Mathcad 2000 is used for this section of Appendix 2.O.

Table 24 Formulas for shear, moment and deflection of flat circular plates of constant thickness

Cases 1a - 1d Annular Plate With Uniform Annular Line Load w at Radius r_o; Outer Edge Simply Supported

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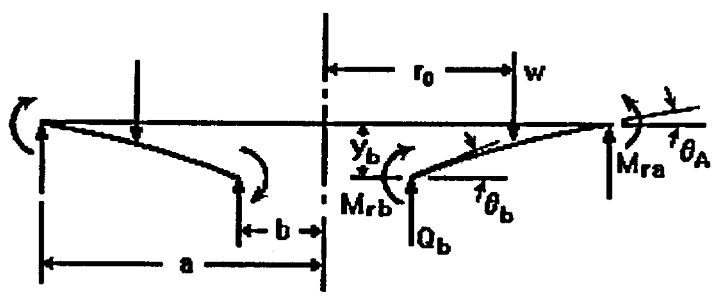
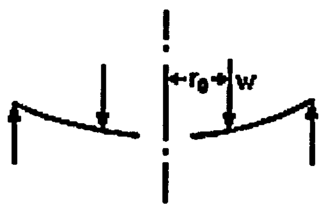
This file corresponds to Cases 1a - 1d in *Roark's Formulas for Stress and Strain.*

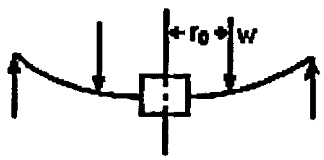
Annular plate with a uniform annular line load w at a radius r_o



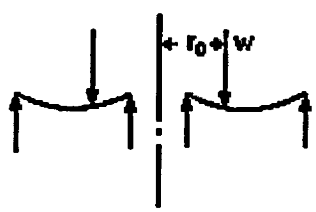
Outer edge simply supported, inner edge free



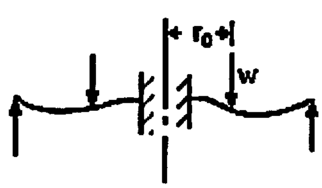
Outer edge simply supported, inner edge guided



Outer edge simply supported, inner edge simply supported



Outer edge simply supported, inner edge fixed



CASE 1A applies to the impact load at the outer periphery. The pipe diameter is the applied load location

Enter dimensions, properties and loading

Plate dimensions:

thickness: $t \equiv 0.75 \cdot in$

outer radius: $a \equiv 2.05 \cdot in$

inner radius: $b \equiv 0.5 \cdot in$

Applied unit load: $w \equiv 9167 \cdot \frac{lbf}{in}$

Modulus of elasticity: $E \equiv 24.625 \cdot 10^6 \cdot \frac{lbf}{in^2}$

Poisson's ratio: $\nu \equiv 0.3$

Radial location of applied load: $r_o \equiv .5 \cdot 3.5 \cdot in$

Constants

Shear modulus: $G \equiv \frac{E}{2 \cdot (1 + \nu)}$

D is a plate constant used in determining boundary values; it is also used in the general equations for deflection, slope, moment and shear. K_{sb} and K_{sro} are tangential shear constants used in determining the deflection due to shear:

$$D \equiv \frac{E \cdot t^3}{12 \cdot (1 - \nu^2)} \qquad D = 9.513 \times 10^5 \, lbf \cdot in$$

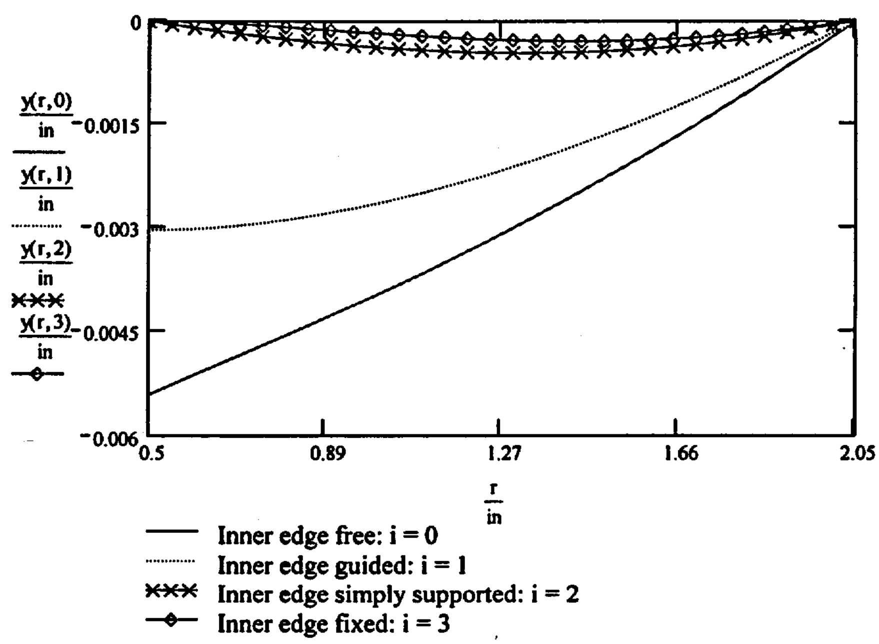
$$K_{sro} \equiv -1.2 \cdot \frac{r_o}{a} \cdot \ln\left(\frac{a}{r_o}\right) \qquad K_{sb} \equiv K_{sro}$$

General formulas and graphs for deflection, slope, moment, shear and stress as a function of r

Define r, the range of the radius and i, the vector index:

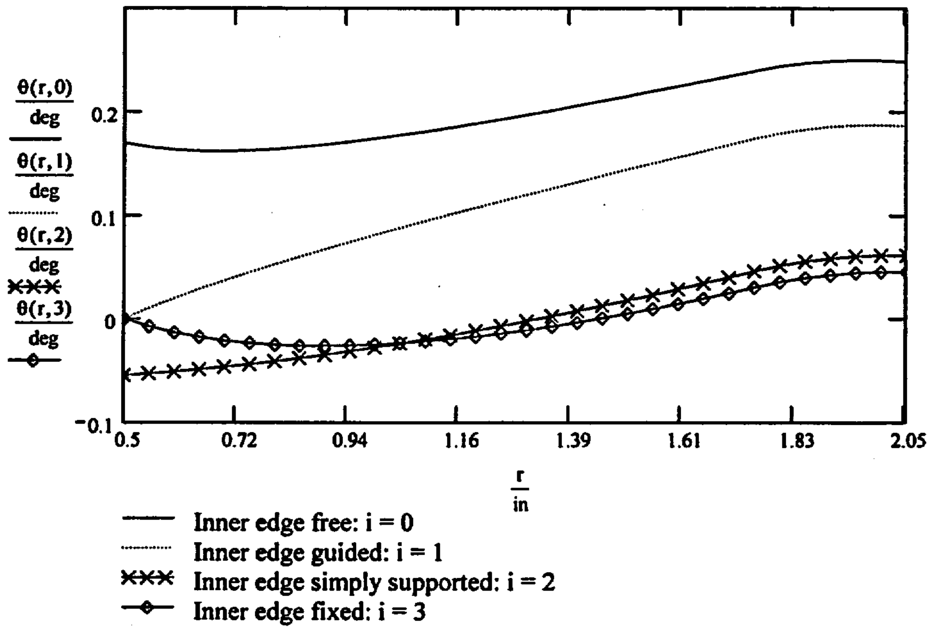
$r \equiv b, 1.1 \cdot b .. a$ $\quad\quad\quad\quad$ $i \equiv 0..3$

Deflection $\quad y(r,i) := y_{b_i} + \theta_{b_i} \cdot r \cdot F_1(r) + M_{rb_i} \cdot \frac{r^2}{D} \cdot F_2(r) + Q_{b_i} \cdot \frac{r^3}{D} \cdot F_3(r) - w \cdot \frac{r^3}{D} \cdot G_3(r)$

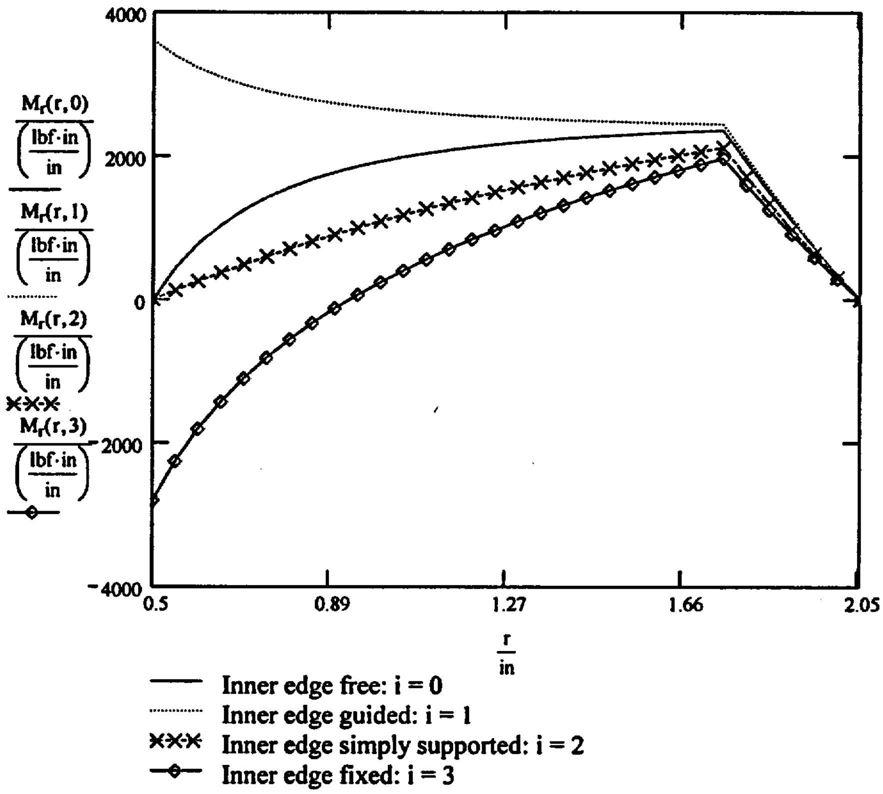


Slope

$$\theta(r,i) := \theta_{b_i}\cdot F_4(r) + M_{rb_i}\cdot\frac{r}{D}\cdot F_5(r) + Q_{b_i}\cdot\frac{r^2}{D}\cdot F_6(r) - w\cdot\frac{r^2}{D}\cdot G_6(r)$$



Radial moment $M_r(r,i) := \theta_{b_i} \cdot \frac{D}{r} \cdot F_7(r) + M_{rb_i} \cdot F_8(r) + Q_{b_i} \cdot r \cdot F_9(r) - w \cdot r \cdot G_9(r)$



The following values are listed in order of inner edge:

- free (i = 0)
- guided (i = 1)
- simply supported (I = 2)
- fixed (I = 3)

Moment at points b and a (inner and outer radius):

$$\frac{M_{rb}}{\left(\frac{lbf \cdot in}{in}\right)} = \begin{pmatrix} 0 \\ 3.595 \times 10^3 \\ 0 \\ -2.798 \times 10^3 \end{pmatrix} \quad \frac{M_{ra}}{\left(\frac{lbf \cdot in}{in}\right)} = \begin{pmatrix} 0 \\ 0 \\ 0 \\ 0 \end{pmatrix}$$

Maximum radial moment (magnitude):

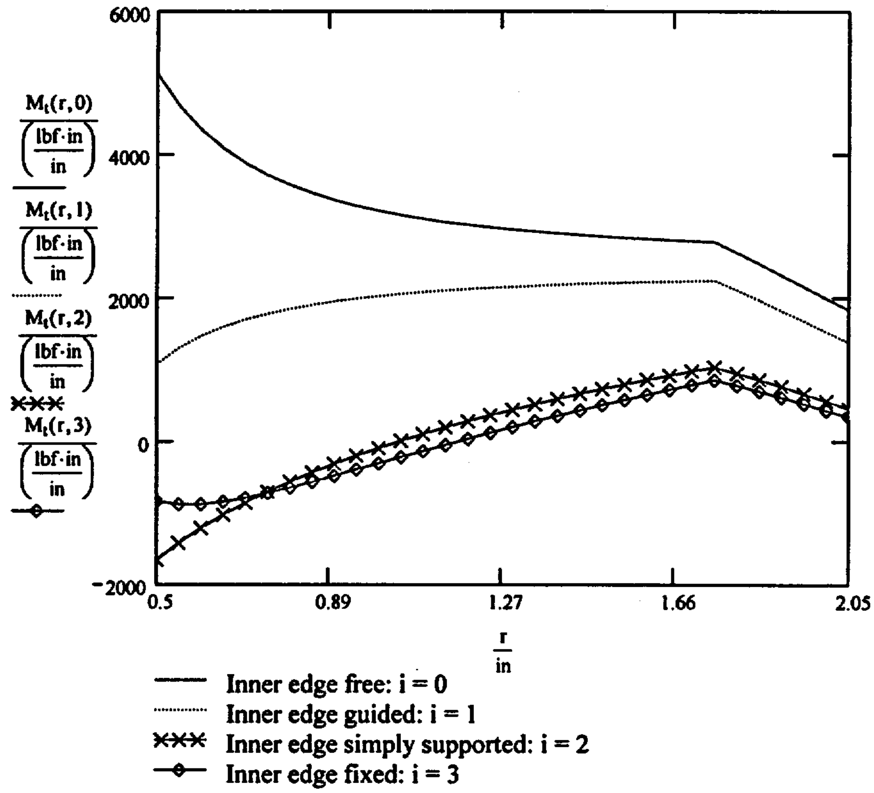
$$Mr_{(r-b)\cdot\frac{100}{in},i} := M_r(r,i) \qquad A_{mr_i} := \max\left(Mr^{\langle i \rangle}\right) \qquad B_{mr_i} := \min\left(Mr^{\langle i \rangle}\right)$$

$$Mr_{max_i} := \left(A_{mr_i} > -B_{mr_i}\right)\cdot A_{mr_i} + \left(A_{mr_i} \le -B_{mr_i}\right)\cdot B_{mr_i}$$

$$\frac{Mr_{max}}{\left(\frac{lbf \cdot in}{in}\right)} = \begin{pmatrix} 2.355 \times 10^3 \\ 3.595 \times 10^3 \\ 2.115 \times 10^3 \\ -2.798 \times 10^3 \end{pmatrix}$$

Transverse moment

$$M_t(r,i) := \frac{\theta(r,i)\cdot D\cdot\left(1-\nu^2\right)}{r} + \nu\cdot M_r(r,i)$$



The following values are listed in order of inner edge:

- **free (i = 0)**
- **guided (i = 1)**
- **simply supported (i = 2)**
- **fixed (i = 3)**

Transverse moment at points b and a (inner and outer radius) due to bending:

$$\frac{M_t(b,i)}{\left(\frac{lbf\cdot in}{in}\right)} = \qquad \frac{M_t(a,i)}{\left(\frac{lbf\cdot in}{in}\right)} =$$

$M_t(b,i)$	$M_t(a,i)$
$5.128\cdot10^3$	$1.828\cdot10^3$
$1.078\cdot10^3$	$1.373\cdot10^3$
$-1.661\cdot10^3$	452.798
-839.265	334.706

Maximum tangential moment (magnitude):

$$Mt_{(r-b)\cdot\frac{100}{in},i} := M_t(r,i) \qquad A_{mt_i} := \max\left(Mt^{\langle i\rangle}\right) \qquad B_{mt_i} := \min\left(Mt^{\langle i\rangle}\right)$$

$$Mt_{max_i} := \left(A_{mt_i} > -B_{mt_i}\right)\cdot A_{mt_i} + \left(A_{mt_i} \le -B_{mt_i}\right)\cdot B_{mt_i}$$

$$\frac{Mt_{max}}{\frac{lbf\cdot in}{in}} = \begin{pmatrix} 5.128\times10^3 \\ 2.234\times10^3 \\ -1.661\times10^3 \\ -884.013 \end{pmatrix} \qquad SF := \frac{M_u}{5128\cdot lbf} \qquad SF = 1.71$$

The remainder of the document displays the general plate functions and constants used in the equations above.

$$C_1 \equiv \frac{1+\nu}{2}\cdot\frac{b}{a}\cdot\ln\left(\frac{a}{b}\right)+\frac{1-\nu}{4}\cdot\left(\frac{a}{b}-\frac{b}{a}\right)$$

$$C_2 \equiv \frac{1}{4}\cdot\left[1-\left(\frac{b}{a}\right)^2\cdot\left(1+2\cdot\ln\left(\frac{a}{b}\right)\right)\right]$$

$$C_3 \equiv \frac{b}{4\cdot a}\cdot\left[\left[\left(\frac{b}{a}\right)^2+1\right]\cdot\ln\left(\frac{a}{b}\right)+\left(\frac{b}{a}\right)^2-1\right]$$

$$C_4 \equiv \frac{1}{2}\cdot\left[(1+\nu)\cdot\frac{b}{a}+(1-\nu)\cdot\frac{a}{b}\right]$$

$$C_5 \equiv \frac{1}{2}\cdot\left[1-\left(\frac{b}{a}\right)^2\right]$$

$$C_6 \equiv \frac{b}{4\cdot a}\cdot\left[\left(\frac{b}{a}\right)^2-1+2\cdot\ln\left(\frac{a}{b}\right)\right]$$

$$C_7 \equiv \frac{1}{2}\cdot\left(1-\nu^2\right)\cdot\left(\frac{a}{b}-\frac{b}{a}\right)$$

$$C_8 \equiv \frac{1}{2}\cdot\left[1+\nu+(1-\nu)\cdot\left(\frac{b}{a}\right)^2\right]$$

$$C_9 \equiv \frac{b}{a}\cdot\left[\frac{1+\nu}{2}\cdot\ln\left(\frac{a}{b}\right)+\left(\frac{1-\nu}{4}\right)\cdot\left[1-\left(\frac{b}{a}\right)^2\right]\right]$$

$$L_3 \equiv \frac{r_o}{4\cdot a}\cdot\left[\left[\left(\frac{r_o}{a}\right)^2+1\right]\cdot\ln\left(\frac{a}{r_o}\right)+\left(\frac{r_o}{a}\right)^2-1\right]$$

$$L_6 \equiv \frac{r_o}{4\cdot a}\cdot\left[\left(\frac{r_o}{a}\right)^2-1+2\cdot\ln\left(\frac{a}{r_o}\right)\right]$$

$$L_9 \equiv \frac{r_o}{a}\cdot\left[\frac{1+\nu}{2}\cdot\ln\left(\frac{a}{r_o}\right)+\frac{1-\nu}{4}\cdot\left[1-\left(\frac{r_o}{a}\right)^2\right]\right]$$

Boundary values due to bending:

At the inner edge of the plate:

$$Q_b \equiv \begin{bmatrix} 0 \cdot \dfrac{lbf}{in} \\ 0 \cdot \dfrac{lbf}{in} \\ w \cdot \left(\dfrac{C_1 \cdot L_9 - C_7 \cdot L_3}{C_1 \cdot C_9 - C_3 \cdot C_7} \right) \\ w \cdot \left(\dfrac{C_2 \cdot L_9 - C_8 \cdot L_3}{C_2 \cdot C_9 - C_3 \cdot C_8} \right) \end{bmatrix} \qquad M_{rb} \equiv \begin{bmatrix} 0 \cdot \dfrac{lbf \cdot in}{in} \\ \dfrac{w \cdot a}{C_8} \cdot L_9 \\ 0 \cdot \dfrac{lbf \cdot in}{in} \\ -w \cdot a \cdot \left(\dfrac{C_3 \cdot L_9 - C_9 \cdot L_3}{C_2 \cdot C_9 - C_3 \cdot C_8} \right) \end{bmatrix}$$

$$y_b \equiv \begin{bmatrix} \dfrac{-w \cdot a^3}{D} \cdot \left(\dfrac{C_1 \cdot L_9}{C_7} - L_3 \right) \\ \dfrac{-w \cdot a^3}{D} \cdot \left(\dfrac{C_2 \cdot L_9}{C_8} - L_3 \right) \\ 0 \cdot in \\ 0 \cdot in \end{bmatrix} \qquad \theta_b \equiv \begin{bmatrix} \dfrac{w \cdot a^2}{D \cdot C_7} \cdot L_9 \\ 0 \cdot deg \\ \dfrac{-w \cdot a^2}{D} \cdot \left(\dfrac{C_3 \cdot L_9 - C_9 \cdot L_3}{C_1 \cdot C_9 - C_3 \cdot C_7} \right) \\ 0 \cdot deg \end{bmatrix}$$

At the outer edge of the plate:

$$y_a \equiv \begin{pmatrix} 0\cdot in \\ 0\cdot in \\ 0\cdot in \\ 0\cdot in \end{pmatrix}$$

$$\theta_a \equiv \begin{bmatrix} \frac{w\cdot a^2}{D}\cdot\left(\frac{C_4\cdot L_9}{C_7} - L_6\right) \\ \frac{w\cdot a^2}{D}\cdot\left(\frac{C_5\cdot L_9}{C_8} - L_6\right) \\ \theta_{b_2}\cdot C_4 + Q_{b_2}\cdot\frac{a^2}{D}\cdot C_6 - \frac{w\cdot a^2}{D}\cdot L_6 \\ M_{rb_3}\cdot\frac{a}{D}\cdot C_5 + Q_{b_3}\cdot\frac{a^2}{D}\cdot C_6 - \frac{w\cdot a^2}{D}\cdot L_6 \end{bmatrix} \qquad Q_a \equiv \begin{pmatrix} -w\cdot\frac{r_o}{a} \\ -w\cdot\frac{r_o}{a} \\ Q_{b_2}\cdot\frac{b}{a} - \frac{w\cdot r_o}{a} \\ Q_{b_3}\cdot\frac{b}{a} - \frac{w\cdot r_o}{a} \end{pmatrix} \qquad M_{ra} \equiv \begin{pmatrix} 0\cdot\frac{lbf\cdot in}{in} \\ 0\cdot\frac{lbf\cdot in}{in} \\ 0\cdot\frac{lbf\cdot in}{in} \\ 0\cdot\frac{lbf\cdot in}{in} \end{pmatrix}$$

Due to tangential shear stresses:

$$y_{sb} \equiv \begin{pmatrix} \frac{K_{sb}\cdot w\cdot a}{t\cdot G} \\ \frac{K_{sb}\cdot w\cdot a}{t\cdot G} \\ 0\cdot in \\ 0\cdot in \end{pmatrix} \qquad y_{sro} \equiv \begin{pmatrix} \frac{K_{sro}\cdot w\cdot r_o}{t\cdot G} \\ \frac{K_{sro}\cdot w\cdot r_o}{t\cdot G} \\ 0\cdot in \\ 0\cdot in \end{pmatrix}$$

$$F_1(r) \equiv \frac{1+\nu}{2}\cdot\frac{b}{r}\cdot\ln\left(\frac{r}{b}\right) + \frac{1-\nu}{4}\cdot\left(\frac{r}{b} - \frac{b}{r}\right) \qquad F_6(r) \equiv \frac{b}{4\cdot r}\cdot\left[\left(\frac{b}{r}\right)^2 - 1 + 2\cdot\ln\left(\frac{r}{b}\right)\right]$$

$$F_2(r) \equiv \frac{1}{4}\cdot\left[1 - \left(\frac{b}{r}\right)^2\cdot\left(1 + 2\cdot\ln\left(\frac{r}{b}\right)\right)\right] \qquad F_7(r) \equiv \frac{1}{2}\cdot\left(1-\nu^2\right)\cdot\left(\frac{r}{b} - \frac{b}{r}\right)$$

$$F_3(r) \equiv \frac{b}{4\cdot r}\cdot\left[\left[\left(\frac{b}{r}\right)^2 + 1\right]\cdot\ln\left(\frac{r}{b}\right) + \left(\frac{b}{r}\right)^2 - 1\right] \qquad F_8(r) \equiv \frac{1}{2}\cdot\left[1 + \nu + (1-\nu)\cdot\left(\frac{b}{r}\right)^2\right]$$

$$F_4(r) \equiv \frac{1}{2}\cdot\left[(1+\nu)\cdot\frac{b}{r} + (1-\nu)\cdot\frac{r}{b}\right] \qquad F_9(r) \equiv \frac{b}{r}\cdot\left[\frac{1+\nu}{2}\cdot\ln\left(\frac{r}{b}\right) + \frac{1-\nu}{4}\cdot\left[1 - \left(\frac{b}{r}\right)^2\right]\right]$$

$$F_5(r) \equiv \frac{1}{2}\cdot\left[1 - \left(\frac{b}{r}\right)^2\right]$$

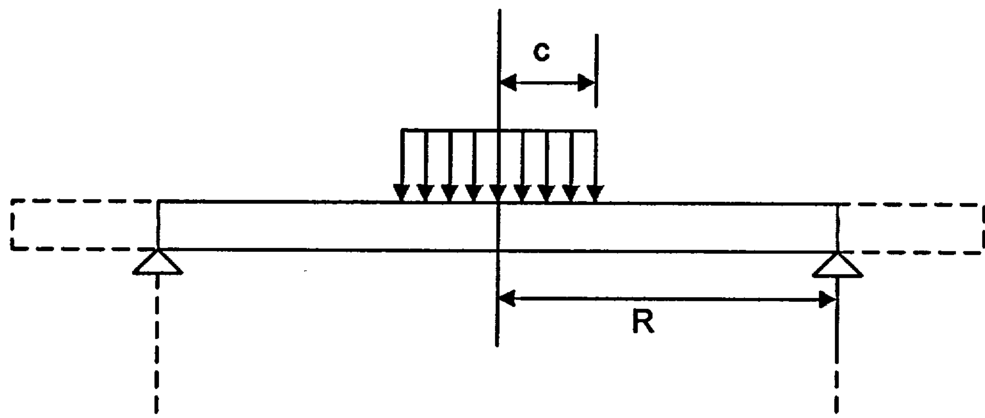
$$G_3(r) \equiv \frac{r_o}{4\cdot r}\cdot\left[\left[\left(\frac{r_o}{r}\right)^2 + 1\right]\cdot\ln\left(\frac{r}{r_o}\right) + \left(\frac{r_o}{r}\right)^2 - 1\right]\cdot(r > r_o)$$

$$G_6(r) \equiv \frac{r_o}{4\cdot r}\cdot\left[\left(\frac{r_o}{r}\right)^2 - 1 + 2\cdot\ln\left(\frac{r}{r_o}\right)\right]\cdot(r > r_o)$$

$$G_9(r) \equiv \frac{r_o}{r}\cdot\left[\frac{1+\nu}{2}\cdot\ln\left(\frac{r}{r_o}\right) + \frac{1-\nu}{4}\cdot\left[1 - \left(\frac{r_o}{r}\right)^2\right]\right]\cdot(r > r_o)$$

The actual safety factor against a complete collapse of the ring like plate is much larger since unlimited large rotations will only occur when a substantial region of the plate has the circumferential moment reach capacity (this can be shown by a limit analysis solution of the plate equations).

The second impact scenario has the loading applied over a region inside the outer diameter of the pipe. To qualify this load case, we consider the plate as simply supported at the pipe diameter and conservatively neglect the overhanging portion of the pipe. Further, we assume the loading is conservatively applied as a uniform pressure over an area equal to the minimum impact diameter of 1.8". For simplicity, we neglect the inner hole in this calculation. Therefore, the limit analysis model for the second impact scenario is shown below:



Calculate effective load area at middle surface assuming a 45 degree spread of load patch

$Hp = 0.75\ in$

$R := 0.5\cdot[(3.5 - 2\cdot 0.226)\cdot in]$

$P = 1.008 \times 10^5\ lbf$

$c := 0.5\cdot(1.8\cdot in + Hp)$

Use inside radius of pipe for this calc.

$M_u = 8.768 \times 10^3\ lbf\cdot\frac{in}{in}$

Using a solution in the text "Introduction to Plasticity"by W. Prager, Addison Wesley, 1959, p. 61, the limit load is

$$P_{lim} := 6\cdot\pi\cdot\frac{M_u}{\left(3 - 2\cdot\frac{c}{R}\right)}$$

Therefore, the safety factor for this case is

$$\frac{P_{lim}}{P} = 1.236$$

Therefore it is concluded that an end plate of diameter and thickness equal to

$Dp = 4.1\ in$ $\qquad$ $Hp = 0.75\ in$

will perform the intended load transfer and limit the movement of the fuel assembly.

APPENDIX 2.R - POCKET TRUNNION RECESS WELD ANALYSIS

2.R.1 Purpose

The purpose of this calculation is to evaluate the pocket trunnion weld stress. There are two bounding cases of interest. The first case is the analysis of the weld under the action of the calculated tie-down loads during transport. Tie-down equations of equilibrium are presented in Section 2.5, and the actual numerical calculations yielding resultant loads are carried out in Appendix 2.C. The second case of interest is an evaluation of the weld (and by implication, the cask material adjacent to the weld) under a condition where the pocket trunnions are subject to 100% of the lifted load. Although the pocket trunnions are not designated as lifting trunnions, it is recognized that they could inadvertantly be subject to such loadings during cask rotation operations. Therefore, to demonstrate the conservatism in the design, the trunnion weld group is assumed subject to the total cask lifted load with an appropriate dynamic load factor and we evaluate whether the material meets the requirements of 10CFR71.45.

2.R.2 Acceptance Criteria

The requirements of CFR71.45 are applied with the calculated stress in the weld group being compared to the material yield strength. Accordingly, under the transport tie down loading, it is required to demonstrate that an acceptable margin of safety exists based on a comparison with the material yield strength. When the trunnion is considered as being subject to the bounding loads during a rotation operation, then an acceptable margin of safety must exist based on a comparison with 33% of the material yield strength.

2.R.3 Assumptions

It is assumed that the weld forces can be computed, using strength of materials theory, based on the weld being a thin line element.

It is assumed that the loads are applied at the center of the trunnion pin for the purpose of computing moment arms.

It is assumed that four intermediate shells provide strength welds.

It is assumed that loads tending to rotate the trunnion with respect to the cask body are resisted by weld forces and by direct bearing on the innermost intermediate shell.

2.R.3 Weld Group Configuration

Figure 2.R.1 shows a sketch of the weld group (view is radially inward toward the center of the overpack. X is along the longitudinal axis of the overpack, Y is vertical, and Z is radially inward. The origin of the coordinate axes is at the centroid of the weld group. Figure 2.R.2 shows a V-groove construction, but a J-groove is also an option.

2.R.4 Weld Group Data Input

The weld is analyzed as a full penetration weld. ASME Section III, Subsection NF-3324.5(f)(3)(d) is the section of the Code that sets the effective weld thickness to be used for such welds. The calculation assumes only 3/4" penetration in the innermost layer

The distance from the root of the weld to the face of the weld is

$$td := 2 \cdot 1.25 \cdot in + 0.75 \cdot in + 1 \cdot in \qquad td = 4.25\,in$$

Therefore, the effective throat thickness for calculation purposes is

$$tw := td \qquad tw = 4.25\,in$$

The length and width of the weld group for calculation purposes are

$$L := 12.375 \cdot in$$

$$w := 12 \cdot in$$

For computation of safety margins, the yield stress of the weld or adjacent base metal is required. For SA-516 Grade 70 material, at 300 degrees F operating temperature, the yield stress is:

$$\sigma_y := 33700 \cdot psi$$

2.R.5 Calculation of Weld Group Area and Moments of Inertia

Weld group property calculations follow the methodology set forth in

Omer Blodgett, Design of Welded Structures, Lincoln Arc Welding Foundation, June 1966.

The weld group is considered as a unit width line element and section properties are computed. Required properties are the area/unit throat thickness, the section modulus/unit throat thickness, and the torsional section modulus/unit throat thickness

Weld Area/unit throat thickness

$$A_{weld} := 2 \cdot L + w \qquad A_{weld} = 36.75\,\text{in}$$

The centroid of the weld group is measured from the edge of the weld group closest to the top end of the overpack.

$$xc := \frac{(2 \cdot L \cdot .5 \cdot L)}{A_{weld}} \qquad xc = 4.167\,\text{in}$$

Now compute the section modulii of the weld group/unit throat thickness around axes through the C.G. (S_{xx} represents the section modulus/unit throat thickness about the longitudinal x axis, for example). Following the reference (Chapter 7, Section 4, Tables 4 and 5

$$S_{xx} := L \cdot w + \frac{w^2}{6} \qquad S_{xx} = 172.5\,\text{in}^2$$

$$S_{yyt} := \frac{\left(2 \cdot w \cdot L + L^2\right)}{3} \qquad S_{yyt} = 150.047\,\text{in}^2$$

$$S_{yyb} := L^2 \cdot \frac{(2 \cdot w + L)}{3 \cdot (w + L)} \qquad S_{yyb} = 76.178\,\text{in}^2$$

The subscripts t and b denote that the property is used when computing forces/unit throat thickness on the top (positive X) section of the weld, or at the ends of the two legs (negative X).

For this trunnion analysis, the use of Syy is extremely conservative since it neglects any resistance in direct compression from the bearing of the underside of the trunnion on the surface of the innermost intermediate shell. To reflect this additional resistance against trunnion rotation about the y axis, we compute the area moment of inertia of the bearing surface between the top leg of the weld and the centoid of the weld group.

$$A_{comp} := xc \cdot w \qquad A_{comp} = 50.005\,in^2$$

$$I_{ycomp} := A_{comp} \cdot \frac{xc^2}{2} + w \cdot \frac{xc^3}{12} \qquad I_{ycomp} = 506.521\,in^4$$

Appropriate adders to the calculated Syy for the weld group alone, to correct for the additional resistance from direct compression on the trunnion bearing surface, are obtained by computing the appropriate section modulii increments per unit of effective throat thickness

$$DS_{yyt} := \frac{I_{ycomp}}{(xc) \cdot tw} \qquad DS_{yyt} = 28.601\,in^2$$

$$DS_{yyb} := \frac{I_{ycomp}}{(L - xc) \cdot tw} \qquad DS_{yyb} = 14.52\,in^2$$

where we have divided the computed section modulus by the weld throat length to conform to the previous definitions. Therefore, the final section modulii for computing the resistance to moments about the y axis are

$$S_{yyt} := S_{yyt} + DS_{yyt} \qquad S_{yyt} = 178.647\,in^2$$

$$S_{yyb} := S_{yyb} + DS_{yyb} \qquad S_{yyb} = 90.698\,in^2$$

The torsional section modulus/unit of throat thickness is

$$J_w := \frac{(w + 2 \cdot L)^3}{12} - L^2 \cdot \frac{(L + w)^2}{(w + 2 \cdot L)} \qquad J_w = 1.66 \times 10^3\,in^3$$

2.R.6 Applied Loading on the Weld Group

The applied loading for any combination is three forces which are assumed to be applied at a given point in space offset from the calculated centroid of the weld. For analysis purposes, the point of application of the loads is defined relative to the extreme forward fiber of the weld group. Defining the offsets as xo, yo and zo, and referencing the pocket trunnion detail drawing for dimensions and Figure 2.R.2, yields:

$$xo := (13 - 4.25)\cdot in$$

xo is computed as the distance from the forward edge of the weld group to the center of the pocket trunnion pin which is assumed located at the center of the half circle which defines the recess cup.

$$xo = 8.75\,in$$

Therefore, the offset in x (longitudinal) is:

$$xoc := xo - xc \qquad xoc = 4.583\,in$$

Using symmetry, yo is determined to be:

$$yo := 0\cdot in$$

zo is the distance along the local z axis from the center of the trunnion pin to the centroid of the weld group. Since four intermediate shells have strength welds, the centroid is at the mid-height of the four intermediate shells. The thicknesses of the five intermediate shells are (numbered from inside out and defined as tg1, tg2, etc.):

$$tg1 := 0.75\cdot in$$

$$tg2 := 1.25\cdot in$$

$$tg3 := tg2$$

$$tg4 := tg2$$

$$tg5 := 1\cdot in$$

The wall thickness of the outer enclosure plate is:

$$t_{wall} := 0.5 \cdot in$$

and the depth of the neutron absorbing material is:

$$depth := 4.4375 \cdot in$$

Therefore, the offset zo is computed as:

$$zo := \left[depth + t_{wall} + .5 \cdot (tg2 + tg3 + tg4 + tg5) - .5 \cdot (3.9375 \cdot in) \right]$$

where half the depth of the recess is used as the location of the load point.

$$zo = 5.344\,in$$

2.R.7 Stress Analysis for Transport Loads

The following load case is limiting for the weld qualification (refer to the summary table in Section 2.5 or the detailed caculation in Appendix 2.C).

$$Fx := 1410000 \cdot lbf$$

$$Fy := 431600 \cdot lbf$$

$$Fr := \left(Fx^2 + Fy^2\right)^{0.5}$$

$$Fr = 1.475 \times 10^6\,lbf$$

Fr is the net direct shear force acting on the weld group. The bending moments on the weld group are determined as:

$$Mx := Fy \cdot zo \qquad Mx = 2.306 \times 10^6\,in \cdot lbf$$

$$My := Fx \cdot zo \qquad My = 7.535 \times 10^6\,lbf \cdot in$$

$$Mz := Fy \cdot xoc \qquad Mz = 1.978 \times 10^6\,in \cdot lbf$$

The direct shear force/unit throat thickness is determined as:

$$S_r := \frac{Fr}{A_{weld}}$$

$$S_r = 4.012 \times 10^4 \frac{lbf}{in}$$

The force/unit throat thickness at point D, which is farthest from the centroid of the weld group, due to the two bending moments is determined as:

$$S_{by} := \frac{Mx}{S_{xx}} \qquad S_{by} = 1.114 \times 10^3 \, ft\,psi$$

$$S_{bx} := \frac{My}{S_{yyb}} \qquad S_{bx} = 8.307 \times 10^4 \frac{lbf}{in}$$

$$S_b := S_{bx} + S_{by} \qquad S_b = 9.644 \times 10^4 \frac{lbf}{in}$$

The torsional force/unit of throat thickness at point D is determined as:

$$S_t := \frac{Mz \cdot \frac{w}{2}}{J_w} \qquad S_t = 7.148 \times 10^3 \frac{lbf}{in}$$

The net force/unit of weld throat thickness is computed as a root mean square

$$S_{eq} := \left[S_b^{\,2} + \left(S_r + S_t\right)^2 \right]^{0.5} \qquad S_{eq} = 1.074 \times 10^5 \frac{lbf}{in}$$

Dividing by the yield strength of the material yields a minimum required throat thickness

$$t_{req} := \frac{S_{eq}}{\sigma_y} \qquad t_{req} = 3.187 \, in$$

and the corresponding safety factor is therefore calculated as:

$$SF := \frac{tw}{t_{req}} \qquad SF = 1.333 \qquad > 0.0$$

2.R.8 Stress Analysis for Rotation Loads

During cask rotation operations, a bounding load for the two pocket trunnions is the total weight of the HI-STAR 100, amplifed by an appropriate inertia load factor. This could potentially occur if there is momentary slack in the liting rig attached to the top lifting trunnions. For this case, an appropriate load set is

$$Fx := \frac{250000}{2} \cdot lbf \qquad Fy := 0 \cdot lbf$$

We note that impact limiters are not installed during this operation.

A 15% dynamic load factor is incorporated into the analysis; therefore,

$$ILF := 1.15$$

Therefore, the amplified longitudinal load on each pocket trunnion is

$$Fx := ILF \cdot Fx \qquad Fr := \left(Fx^2 + Fy^2\right)^{0.5}$$

$$Fr = 1.438 \times 10^5 \, lbf$$

Fr is the net direct in-plane force acting on the weld group. The bending moments on the weld group are determined as:

$$Mx := Fy \cdot zo \qquad Mx = 0 \, in \cdot lbf$$

$$My := Fx \cdot zo \qquad My = 7.682 \times 10^5 \, lbf \cdot in$$

$$Mz := Fy \cdot xoc \qquad Mz = 0 \, in \cdot lbf$$

The direct shear force/unit throat thickness is determined as:

$$S_r := \frac{Fr}{A_{weld}} \qquad S_r = 3.912 \times 10^3 \frac{lbf}{in}$$

The force/unit throat thickness at point D, which is farthest from the centroid of the weld group, due to the two bending moments is determined as:

$$S_{by} := \frac{Mx}{S_{xx}} \qquad S_{by} = 0\,ft\,psi$$

$$S_{bx} := \frac{My}{S_{yyb}} \qquad S_{bx} = 8.469 \times 10^3 \frac{lbf}{in}$$

$$S_b := S_{bx} + S_{by} \qquad S_b = 8.469 \times 10^3 \frac{lbf}{in}$$

The torsional force/unit of throat thickness at point D is determined as:

$$S_t := \frac{Mz \cdot \frac{w}{2}}{J_w} \qquad S_t = 0 \frac{lbf}{in}$$

The net force/unit of weld throat thickness is computed as a root mean square

$$S_{eq} := \left[S_b^{\,2} + \left(S_r + S_t\right)^2 \right]^{0.5} \qquad S_{eq} = 9.329 \times 10^3 \frac{lbf}{in}$$

Dividing by 33% of the material yield strength to meet the limits inferred by 10CFR71.45 for cask stress during a "lifting" operation gives a minimum required throat thickness

$$t_{req} := \frac{S_{eq}}{.333 \cdot \sigma_y} \qquad t_{req} = 0.831\,in$$

so that the corresponding safety margin for this case is calculated as:

$$SF := \frac{tw}{t_{req}} \qquad SF = 5.112 \qquad > 0.0$$

Model the weld as a 3/16" groove weld for calculation purposes.

$$t_{weld} := \frac{3}{16} \cdot in$$

The shear stress due to the internal pressure in the vertical flat panel weld is

$$\tau_{weld} := \frac{q \cdot .5 \cdot b_s}{t_{weld}} \qquad \tau_{weld} = 630\,psi$$

Since the allowable base metal stress for primary bending has been input earlier, we divide this value by 1.5 to obtain an allowable primary membrane stress.

$$SF := \frac{.6 \cdot \left(\frac{S_a}{1.5}\right)}{\tau_{weld}} \qquad SF = 16.698$$

For the weld around the annular ring, we note that since the unsupported strip width is less than the value used above, the weld shear stress will be even lower. Thus, the flat panel weld controls the design.

2.AM.4 Conclusion

For a 30 psi internal pressure, all safety factors are well in excess of 1.0 demonstrating that the 30 psig internal pressure is safely supported by the enclosure shell and the enclosure shell return.

Although the effect of dead weight of the neutron absorber material has not been included as an additional loading in the analysis of the enclosure shell return, it is clear from the large safety factors that structural integrity will not be compromised.

There is no credible mechanism for the pressure to exceed 30 psi under normal operating conditions in the enclosure shell sectors.

APPENDIX 2.AO - ANALYSIS OF TRANSNUCLEAR DAMAGED FUEL CANISTER AND THORIA ROD CANISTER

2.AO.1 Introduction

Some of the items at the Dresden Station that have been considered for transport in the HI-STAR 100 System are damaged fuel stored in Transnuclear damaged fuel canisters and Thoria rods that are also stored in a special canister designed by Transnuclear. Both of these canisters have been designed and have been used by ComEd to transport the damaged fuel and the Thoria rods. Despite the previous usage of these canisters, it is prudent and appropriate to provide an independent structural analysis of the major load path of these canisters prior to accepting them for inclusion as permitted items in the HI-STAR 100 MPC's. This appendix contains the necessary structural analysis of the Transnuclear damaged fuel canister and Thoria rod canister. The objective of the analysis is to demonstrate that the canisters are structurally adequate to support the loads that develop during normal lifting operations and during postulated hypothetical accident conditions.

The upper closure assembly is designed to meet the requirements of NUREG-0612 [2]. The remaining components of the canisters are governed by ASME Code Section III, Subsection NG [3]. These are the same criteria used in Appendix 2.AH to analyze the Holtec damaged fuel container for Dresden damaged fuel.

2.AO.2 Composition

This appendix was created using the Mathcad (version 8.02) software package. Mathcad uses the symbol ':=' as an assignment operator, and the equals symbol '=' retrieves values for constants or variables.

2.AO.3 References

1. Crane Manufacture's of America Association, Specifications for Electric Overhead Traveling Cranes #70.

2. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants

3. ASME Boiler and Pressure Vessel Code, Section III, July 1995

2.AO.4 Assumptions

1. Buckling is not a concern during an accident since during a drop the canister will be confined by the fuel basket.

2. The strength of the weld is assumed to decrease the same as the base metal as the temperature increases.

2.AO.5 Method

Two are considered: 1) normal lifting and handling of canister, and 2) hypothetical accident drop event.

2.AO.6 Acceptance Criteria

1) Normal Handling -

a) Canister governed by ASME NG allowables:

b)Welds governed by NG and NF allowables;
quality factors taken from NG
stress limit = 0.3 Su

c) Lifting governed by NUREG-0612 allowables.

2) Drop Accident -

a) canister governed by ASME NG allowables:
shear = 0.42 Su (conservative)

b)Welds governed by NG and NF allowables;
quality factors taken from NG
stress limit = 0.42 Su

2.AO.7 Input Stress Data

The canisters is handled while still in the spent fuel pool. Therefore, its design temperature for lifting considerations is the temperature of the fuel pool water (150°F). The design temperature for accident conditions is 725°F. All dimensions are taken from the Transnuclear design drawings listed at the end of this appendix. The basic input parameters used to perform the calculations are:

Design stress intensity of SA240-304 (150°F)	$S_{m1} := 20000 \cdot psi$
Design stress intensity of SA240-304 (775°F)	$S_{m2} := 15800 \cdot psi$
Yield stress of SA240-304 (150°F)	$S_{y1} := 27500 \cdot psi$
Yield stress of SA240-304 (775°F)	$S_{y2} := 17500 \cdot psi$
Ultimate strength of SA240-304 (150°F)	$S_{u1} := 73000 \cdot psi$
Ultimate strength of SA240-304 (775°F)	$S_{u2} := 63300 \cdot psi$

Ultimate strength of weld material (150°F)	$Su_w := 70000 \cdot psi$
Ultimate strength of weld material (775°F)	$Su_{wacc} := Su_w - (S_{u1} - S_{u2})$
Weight of a BWR fuel assembly (D-1)	$W_{fuel} := 400 \cdot lbf$
Weight of 18 Thoria Rods (Calculated by Holtec)	$W_{thoria} := 90 \cdot lbf$
Bounding Weight of the damaged fuel canister (Estimated by Holtec)	$W_{container} := 150 \cdot lbf$
Bounding Weight of the Thoria Rod Canister (Estimated)	$W_{rodcan} := 300 \cdot lbf$
Quality factor for full penetration weld (visual inspection)	$n := 0.5$
Dynamic load factor for lifting	$DLF := 1.15$

The remaining input data is provided as needed in the calculation section

2.AO.8 Calculations for Transnuclear Damaged Fuel Canister

2.AO.8.1 Lifting Operation (Normal Condition of Storage)

The critical load case under normal conditions is the lifting operation. The key areas of concern for ASME NG analysis are the canister sleeve, the sleeve to lid frame weld, and the lid frame. All calculations performed for the lifting operation assume a dynamic load factor of 1.15 [1].

2.AO.8.1.1 Canister Sleeve

During a lift, the canister sleeve is loaded axially, and the stress state is pure tensile membrane. For the subsequent stress calculation, it is assumed that the full weight of the damaged fuel canister and the fuel assembly are supported by the sleeve. The magnitude of the load is

$$F := DLF \cdot (W_{container} + W_{fuel}) \qquad F = 632 \cdot lbf$$

From TN drawing 9317.1-120-4, the canister sleeve geometry is

$id_{sleeve} := 4.81 \cdot in \qquad t_{sleeve} := 0.11 \cdot in$

The cross sectional area of the sleeve is

$$A_{sleeve} := (id_{sleeve} + 2 \cdot t_{sleeve})^2 - id_{sleeve}^2 \qquad A_{sleeve} = 2.16 \cdot in^2$$

Therefore, the tensile stress in the sleeve is $\sigma := \frac{F}{A_{sleeve}}$ $\sigma = 292 \cdot psi$

The allowable stress intensity for the primary membrane category is S_m per Subsection NG of the ASME Code. The corresponding safety margin is

$$SM := \frac{S_{m1}}{\sigma} - 1 \qquad SM = 67.5$$

2.AO.8.1.2 Sleeve Welds

The top of the canister must support the amplified weight. This load is carried directly by the fillet weld that connects the lid frame to the canister sleeve. The magnitude of the load is conservatively taken a the entire amplified weight of canister plus fuel.

$$F = 632 \cdot lbf$$

The weld thickness is $t_{base} := 0.09 \cdot in$

The area of the weld, with proper consideration of quality factors, is

$$A_{weld} := n \cdot 4 \cdot \left(id_{sleeve} + 2 \cdot t_{sleeve}\right) \cdot .7071 \cdot t_{base} \qquad A_{weld} = 0.64 \cdot in^2$$

Therefore, the shear stress in the weld is $\tau := \frac{F}{A_{weld}}$ $\tau = 988 \cdot psi$

From the ASME Code the allowable weld shear stress, under normal conditions (Level A), is 30% of the ultimate strength of the base metal. The corresponding safety margin is

$$SM := \frac{0.3 \cdot S_{u1}}{\tau} - 1 \qquad SM = 21.2$$

2.AO.8.1.3 Lid Frame Assembly

The Lid Frame assembly is classified as a NUREG-0612 lifting device. As such the allowable stress for design is the lesser of one-sixth of the yield stress and one-tenth of the ultimate strength.

$$\sigma_1 := \frac{S_{y1}}{6} \qquad \sigma_2 := \frac{S_{u1}}{10}$$

$$\sigma_1 = 4583 \cdot psi \qquad \sigma_2 = 7300 \cdot psi$$

For SA240-304 material the yield stress governs. $\sigma_{allowable} := \sigma_1$

The total lifted load is $F := DLF \cdot (W_{container} + W_{fuel})$ $F = 632 \cdot lbf$

The frame thickness is obtained from Transnuclear drawing 9317.1-120-11

$t_{frame} := 0.395 \cdot in$

The inside span is the same as the canister sleeve $id_{sleeve} = 4.81 \cdot in$

The area available for direct load is

$A_{frame} := (id_{sleeve} + 2 \cdot t_{frame})^2 - id_{sleeve}^2$ $A_{frame} = 8.224 \cdot in^2$

The direct stress in the frame is

$\sigma := \dfrac{F}{A_{frame}}$ $\sigma = 77 \cdot psi$

The safety margin is

$SM := \dfrac{\sigma_{allowable}}{\sigma} - 1$ $SM = 58.59$

The bearing stress at the four lift locations is computed from the same drawing

$A_{bearing} := 4 \cdot t_{frame} \cdot (2 \cdot 0.38 \cdot in)$ $A_{bearing} = 1.201 \cdot in^2$

$\sigma_{bearing} := \dfrac{F}{A_{bearing}}$ $\sigma_{bearing} = 526.732 \cdot psi$ $SM := \dfrac{\sigma_{allowable}}{\sigma_{bearing}} - 1$ $SM = 7.7$

2.AO.8.2 60g End Drop (Hypothetical Accident Condition of Transport)

The critical member of the damaged fuel canister during the end drop scenario is the bottom assembly (see Transnuclear drawing 9317.1-120-5). It is subjected to direct compression due to the amplified weight of the fuel assembly and the canister. The bottom assembly is a 3.5" Schedule 40S pipe. The load due to the 60g end drop is

$F := 60 \cdot (W_{fuel} + W_{container})$ $F = 33000 \cdot lbf$

The properties of the pipe are obtained from the Ryerson Stock Catalog as

$od := 4 \cdot in$ $id := 3.548 \cdot in$ $t_{pipe} := \dfrac{(od - id)}{2}$ $t_{pipe} = 0.226 \cdot in$

The pipe area is

$$A_{pipe} := \frac{\pi}{4} \cdot \left(od^2 - id^2\right) \qquad A_{pipe} = 2.68 \cdot in^2$$

The stress in the member is

$$\sigma := \frac{F}{A_{pipe}} \qquad \sigma = 12316 \cdot psi$$

The allowable primary membrane stress from Subsection NG of the ASME Code, for accident conditions (Level D), is

$$\sigma_{allowable} := 2.4 \cdot S_{m2} \qquad \sigma_{allowable} = 37920 \cdot psi$$

The safety margin is

$$SM := \frac{\sigma_{allowable}}{\sigma} - 1 \qquad SM = 2.1$$

To check the stability of the pipe, we conservatively compute the Euler Buckling load for a simply supported beam.

The Young's Modulus is

$$E := 27600000 \cdot psi$$

Compute the moment of inertia as

$$I := \frac{\pi}{64} \cdot \left(od^4 - id^4\right) \qquad I = 4.788 \cdot in^4$$

$$L := 22 \cdot in \qquad P_{crit} := \pi^2 \cdot \frac{E \cdot I}{L^2} \qquad P_{crit} = 2.695 \cdot 10^6 \cdot lbf$$

The safety margin is

$$SM := \frac{P_{crit}}{F} - 1 \qquad SM = 80.654$$

2.AO.8.3 Conclusion for TN Damaged Fuel Canister

The damaged fuel canister and the upper closure assembly are structurally adequate to withstand the specified normal and accident condition loads. All calculated safety margins are greater than zero.

2.AO.9 Calculations for Transnuclear Thoria Rod Canister

2.AO.9.1 Lifting Operation (Normal Condition of Storage)

The critical load case under normal conditions is the lifting operation. The key areas of concern for ASME NG analysis are the canister sleeve, the sleeve to lid frame weld, and the lid frame. All calculations performed for the lifting operation assume a dynamic load factor of 1.15.

2.AO.9.1.1 Canister Sleeve

During a lift, the canister sleeve is loaded axially, and the stress state is pure tensile membrane. For the subsequent stress calculation, it is assumed that the full weight of the Thoria rod canister and the Thoria rods are supported by the sleeve. The magnitude of the load is

$$F := DLF \cdot (W_{rodcan} + W_{thoria}) \qquad F = 449 \cdot lbf$$

From TN drawing 9317.1-182-1, the canister sleeve geometry is

$id_{sleeve} := 4.81 \cdot in \qquad t_{sleeve} := 0.11 \cdot in$

The cross sectional area of the sleeve is

$$A_{sleeve} := (id_{sleeve} + 2 \cdot t_{sleeve})^2 - id_{sleeve}^2 \qquad A_{sleeve} = 2.16 \cdot in^2$$

Therefore, the tensile stress in the sleeve is $\sigma := \dfrac{F}{A_{sleeve}}$ $\qquad \sigma = 207 \cdot psi$

The allowable stress intensity for the primary membrane category is S_m per Subsection NG of the ASME Code. The corresponding safety margin is

$$SM := \frac{S_{m1}}{\sigma} - 1 \qquad SM = 95.5$$

2.AO.9.1.2 Sleeve Welds

The top of the canister must support the amplified weight. This load is carried directly by the fillet weld that connects the lid frame to the canister sleeve. The magnitude of the load is conservatively taken a the entire amplified weight of canister plus Thoria rod.

$$F = 449 \cdot lbf$$

The weld thickness is $t_{base} := 0.09 \cdot in$ (assumed equal to the same weld for the damaged fuel canister)

The area of the weld, with proper consideration of quality factors, is

$$A_{weld} := n \cdot 4 \cdot (id_{sleeve} + 2 \cdot t_{sleeve}) \cdot .7071 \cdot t_{base} \qquad A_{weld} = 0.64 \cdot in^2$$

Therefore, the shear stress in the weld is $\tau := \dfrac{F}{A_{weld}}$ $\qquad \tau = 701 \cdot psi$

From the ASME Code the allowable weld shear stress, under normal conditions (Level A), is 30% of the ultimate strength of the base metal. The corresponding safety margin is

$$SM := \frac{0.3 \cdot S_{u1}}{\tau} - 1 \qquad SM = 30.3$$

2.AO.9.1.3 Lid Frame Assembly

The Lid Frame assembly is classified as a NUREG-0612 lifting device. As such the allowable stress for design is the lesser of one-sixth of the yield stress and one-tenth of the ultimate strength.

$$\sigma_1 := \frac{S_{y1}}{6} \qquad \sigma_2 := \frac{S_{u1}}{10}$$

$$\sigma_1 = 4583 \cdot psi \qquad \sigma_2 = 7300 \cdot psi$$

For SA240-304 material the yield stress governs. $\sigma_{allowable} := \sigma_1$

The total lifted load is $F := DLF \cdot (W_{rodcan} + W_{thoria})$ $F = 449 \cdot lbf$

The frame thickness is obtained from Transnuclear drawing 9317.1-182-8. This drawing was not available, but the TN drawing 9317.1-182-4 that included a view of the lid assembly suggests that it is identical in its structural aspects to the lid frame in the damaged fuel canister.

$$t_{frame} := 0.395 \cdot in$$

The inside span is the same as the canister sleeve $id_{sleeve} = 4.81 \cdot in$

The area available for direct load is

$$A_{frame} := (id_{sleeve} + 2 \cdot t_{frame})^2 - id_{sleeve}^2 \qquad A_{frame} = 8.224 \cdot in^2$$

The direct stress in the frame is

$$\sigma := \frac{F}{A_{frame}} \qquad \sigma = 55 \cdot psi$$

The safety margin is

$$SM := \frac{\sigma_{allowable}}{\sigma} - 1 \qquad SM = 83.04$$

The bearing stress at the four lift locations is computed from the same drawing

$$A_{bearing} := 4 \cdot t_{frame} \cdot (2 \cdot 0.38 \cdot in) \qquad A_{bearing} = 1.201 \cdot in^2$$

$$\sigma_{bearing} := \frac{F}{A_{bearing}} \qquad \sigma_{bearing} = 373.501 \cdot psi \qquad SM := \frac{\sigma_{allowable}}{\sigma_{bearing}} - 1 \qquad SM = 11.27$$

2.AO.9.2 60g End Drop (Hypothetical Accident Condition of Transport)

The critical member of the damaged fuel canister during the drop scenario is the bottom assembly. Transnuclear drawing 9317.1-120-5). It is subjected to direct compression due to the amplified weight of the Thoria rods and the canister.

$F := 60 \cdot (W_{thoria} + W_{rodcan})$ $\qquad F = 23400 \cdot lbf$

The properties of the pipe are obtained from the Ryerson Stock Catalog as

$od := 4 \cdot in$ $\qquad id := 3.548 \cdot in$ $\qquad t_{pipe} := \frac{(od - id)}{2}$ $\qquad t_{pipe} = 0.226 \cdot in$

The pipe area is $\qquad A_{pipe} := \frac{\pi}{4} \cdot (od^2 - id^2)$ $\qquad A_{pipe} = 2.68 \cdot in^2$

The stress in the member is $\qquad \sigma := \frac{F}{A_{pipe}}$ $\qquad \sigma = 8733 \cdot psi$

The allowable primary membrane stress from Subsection NG of the ASME Code, for accident conditions (Level D), is

$\sigma_{allowable} := 2.4 \cdot S_{m2}$ $\qquad \sigma_{allowable} = 37920 \cdot psi$

The safety margin is $\qquad SM := \frac{\sigma_{allowable}}{\sigma} - 1$ $\qquad SM = 3.3$

To check the stability of the pipe, we compute the Euler Buckling load for a simply supported beam.

The Young's Modulus is $\qquad E := 27600000 \cdot psi$

Compute the moment of inertia as $\qquad I := \frac{\pi}{64} \cdot (od^4 - id^4)$ $\qquad I = 4.788 \cdot in^4$

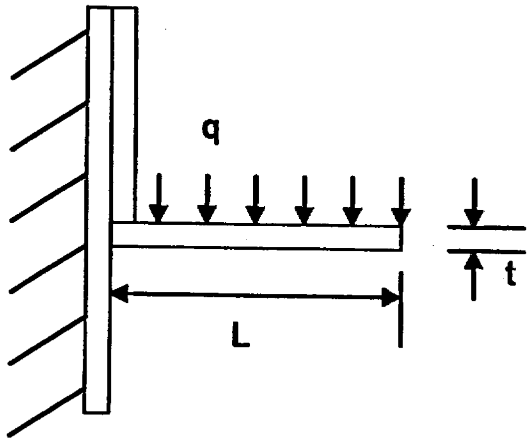
$L := 22 \cdot in$

$P_{crit} := \pi^2 \cdot \frac{E \cdot I}{L^2}$ $\qquad P_{crit} = 2.695 \cdot 10^6 \cdot lbf$

The safety margin is $\qquad SM := \frac{P_{crit}}{F} - 1$ $\qquad SM = 114.153$

2.AO.9.4 60g Side Drop (Hypothetcal Accident Condition of Transport)

The Thoria Rod Separator Assembly is shown in TN drawings 9317.1-182-1 and 9317.1-182-3. Under the design basis side drop, we examine the consequences to one of the rod support strips acting as a cantilever strip acted upon by self-weight and the weight of one Thoria rod.



Weight of 1 rod per unit length

$length := 113.16 \cdot in$

$w_{rod} := 90 \cdot \frac{lbf}{18} \cdot \frac{1}{length}$ $\qquad w_{rod} = 0.044 \cdot \frac{lbf}{in}$

Weight of support per unit length (per drawing 9317.1-182-3

$L := 1.06 \cdot in$ $\qquad t := 0.11 \cdot in$

$w_{sup} := .29 \cdot \frac{lbf}{in^3} \cdot L \cdot t$ $\qquad w_{sup} = 0.034 \cdot \frac{lbf}{in}$

Amplified load (assumed as a uniform distribution)

$q := 60 \cdot \left(w_{rod} + w_{sup}\right)$ $\qquad q = 4.68 \cdot \frac{lbf}{in}$

$Moment := \frac{q \cdot L^2}{2}$ $\qquad Moment = 2.629 \cdot in \cdot lbf$

Bending stress at the root of the cantilever beam is

$\sigma := 6 \cdot \frac{Moment}{1 \cdot in \cdot t^2}$ $\qquad \sigma = 1.304 \cdot 10^3 \cdot psi$

Shear stress at the root of the cantilever $\qquad \tau := q \cdot \frac{L}{t \cdot 1 \cdot in}$ $\qquad \tau = 45.098 \cdot psi$

Large margins of safety are indicated by these stress results.

2.AO.9.5 Conclusion for TN Thoria Rod Canister

The Thoria rod canister is structurally adequate to withstand the specified normal lift and hypothetical accident condition loads. All calculated safety margins are greater than zero.

2.AO.10 General Conclusion

The analysis of the TN damaged fuel canister and the TN Thoria rod canister have demonstrated that all structural safety margins are large. We have confirmed that the TN canisters have positive safety margins for the HI-STAR 100 govening design basis loads. Therefore, the loaded TN canisters from ComEd Dresden Unit#1 can safely be carried in the HI-STAR 100 System.

2.AO.11 List of Transnuclear Drawing Numbers

9317.1-120 - 2,3,4,5,6,7,8,9,10,11,13,14,15,17,18,19,20,21,22,23

9317.1-182- 1,2,3,4,5,6

Table 3.3.2

SUMMARY OF PWR ASSEMBLY RODS INITIAL FILL GAS DATA

Assembly Type	Rods Per Assembly	Free Rod Volume (inch^3)	Fill Pressure (psig)	Maximum Fill Volume (Liters at STP†)	
				Per Rod	Per Assembly
W 14×14 Std.	179	1.72	0-460	0.845	151.2
W 15×15 Std.	204	1.25	0-475	0.633	129.1
W 17×17 Std.	264	1.05-1.25	275-500	0.666	175.8
B&W 15×15 Mark B	208	1.308	415	0.582	121.1
B&W 17×17 Mark C	264	0.819	435	0.381	100.6
CE 14×14 Std.	164	1.693	300-450	0.814	133.5
CE 16×16 Sys 80	220	1.411	300-450	0.678	149.2
B&W-15x15 Mark B-11	*208*	*1.260*	*415*	*0.524*	*109.0*
CE-14x14 (MP2)	*176*	*1.728*	*300-450*	*0.777*	*136.8*

† STP stands for standard temperature and pressure.

Table 3.3.3

BOUNDING VALUES OF FUEL CLADDING STRESS FOR PWR SNF

	W 14×14 Std.	W 15×15 Std.	W 17×17 Std.	B&W 15×15 Mark B	B&W 17×17 Mark C	CE 14×14 Std.	CE 16×16 Sys 80	*B&W 15x15 Mark B-11*	*CE 14x14 (MP2)*
Fresh Fuel Rods O.D. (inch)	0.4220	0.422	0.374	0.430	0.379	0.440	0.382	*0.414*	*0.440*
End of Life Oxidation Thickness (inch)†	0.0027	0.0027	0.0027	0.0027	0.0027	0.0027	0.0027	*0.0027*	*0.0027*
End of Life Rods O.D. (inch)	0.4166	0.4166	0.3686	0.4246	0.3736	0.4346	0.3766	*0.4086*	*0.4346*
Rods I.D. (inch)	0.3734	0.373	0.329	0.377	0.331	0.384	0.332	*0.370*	*0.388*
Average tube Diameter (inch)	0.3950	0.3948	0.3488	0.4008	0.3523	0.4093	0.3493	*0.3893*	*0.4113*
Wall Thickness (inch)	0.0216	0.0218	0.0198	0.0238	0.0213	0.0253	0.0223	*0.0193*	*0.0233*
Hot Volume Pressure at 300°C (MPa)††	9.77	10.67	10.08	9.62	10.87	10.01	9.61	*9.40*	*9.23*
Cladding Stress (MPa)	89.3	96.7	88.8	81.0	90.0	81.0	75.2	*94.8*	*81.4*

† PNL-4835 [3.3.5] reported maximum cladding thickness loss due to in-reactor oxidation.

†† This average rod gas temperature conservatively bounds the plenum gas temperature.

Table 3.3.5

SUMMARY OF BWR ASSEMBLY RODS INITIAL GAS FILL DATA

Assembly Type	Rods Per Assembly	Free Rod Volume ($inch^3$)	Fill Pressure (psig)	Max. Fill Volume (Liters at STP)	
				Per Rod	Per Assy
GE 7×7 (1966)	49	2.073	0-44.1†	0.126	6.17
GE 7×7 (1968)	49	2.073	0-44.1	0.126	6.17
GE 7×7R	49	1.991	0-44.1	0.121	5.93
GE 8×8	60	1.504	0-44.1	0.0915	5.49
GE 8×8R	60	1.433	0-147††	0.240	14.4
Exxon 9×9	79	1.323	58.8-88.2†††	0.141	11.1
6×6 GE Dresden-1	36	2.304	58.8-88.2	0.245	8.82
6×6 GE Dresden-1 MOX	36	2.286	58.8-88.2	0.243	8.75
6×6 GE Humboldt Bay	36	2.346	58.8-88.2	0.250	9.0
7×7 GE Humboldt Bay	49	*1.666* ~~1.662~~	58.8-88.2	0.177	8.67
8×8 GE Dresden-1	64	1.235	58.8-88.2	0.131	8.38
8×8 SPC	63	1.615	58.8-88.2	0.172	10.8
9×9 SPC w/2 water rods	79	1.248	58.8-88.2	0.133	10.5
9×9 SPC w/1 water rod	80	1.248	58.8-88.2	0.133	10.6
9×9 GE11/GE13	74	1.389	58.8-88.2	0.150	11.1
9×9 Atrium 9B SPC	72	1.366	58.8-88.2	0.145	10.4
10×10 SVEA-96	96	1.022	58.8-88.2	0.109	10.5
10×10 GE12/GE14	92	1.167	58.8-88.2	0.124	11.4
6x6 Dresden Thin Und	*36*	*2.455*	*58.8-88.2*	*0.261*	*9.4*
7x7 Oyster Creek	*49*	*2.346*	*58.8-88.2*	*0.234*	*11.5*

† Conservatively bounding for GE-7x7 (1966), GE-7x7 (1968), GE-7x7R and GE-8x8 (ORNL/TM-9591/V1-R1).

†† Conservatively bounding for GE-8x8R (ORNL/TM-9591/V1-R1 reports 3 atm).

††† BWR fuel rods internal pressurization between 4 and 6 atm (PNL-4835).

Table 3.3.5 (continued) SUMMARY OF BWR ASSEMBLY RODS INITIAL GAS FILL DATA					
Assembly Type	**Rods Per Assembly**	**Free Rod Volume (inch^3)**	**Fill Pressure (psig)**	**Max. Fill Volume (Liters at STP)**	
				Per Rod	**Per Assy**
8x8 Oyster Creek	*64*	*1.739*	*58.8-88.2*	*0.173*	*11.1*
8x8 Quad†	*64*	*1.201*	*58.8-88.2*	*0.120*	*7.68*
8x8 TVA Browns Ferry	*61*	*1.686*	*58.8-88.2*	*0.168*	*10.2*
9x9 SPC-5	*76*	*1.249*	*58.8-88.2*	*0.124*	*9.4*

Table 3.3.6

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNF

Fuel Type	Fresh Fuel Rod O.D. (in.)	End of Life Rod O.D.† (in.)	Rod I.D. (in.)	Avg. Tube Diameter (in.)	Wall Thickness (in.)	Hot Vol. Pressure at 300°C (MPa)	Cladding Stress (MPa)
GE 7×7 (1966)	0.563	0.5536	0.499	0.5263	0.0273	4.61	44.4
GE 7×7 (1968)	0.570	0.5606	0.499	0.5298	0.0308	4.61	39.6
GE 7×7R	0.563	0.5536	0.489	0.5213	0.0323	4.76	38.4
GE 8×8	0.493	0.4836	0.425	0.4543	0.0293	5.08	39.4
GE 8×8R	0.483	0.4736	0.419	0.4463	0.0273	6.68	54.7
Exxon 9×9	0.42	0.4106	0.36	0.3853	0.0253	5.08	38.7
6×6 GE Dresden-1	0.5645	0.5551	0.4945	0.5248	0.0303	6.1	52.8
6×6 MOX Dresden-1	0.5625	0.5531	0.4925	0.5228	0.0303	6.1	52.8
Humboldt Bay 6×6	0.563	0.5536	0.499	0.5263	0.0273	5.98	57.6††
Humboldt Bay 7×7	0.486	0.4766	~~0.42~~ *0.4204*	~~0.4483~~ *0.4485*	~~0.0283~~ *0.0281*	~~6.15~~ *6.13*	~~48.7~~ *48.9*
8×8 GE Dresden-1	0.412	0.4026	0.362	0.3813	0.0203	6.29	59.1††

† Excludes 0.0047 inch end of life oxidation thickness.

†† These fuel types are separately evaluated for peak fuel cladding temperature limits.

Table 3.3.6 (continued)

BOUNDING VALUES OF FUEL CLADDING STRESS FOR BWR SNF

Fuel Type	Fresh Fuel Rod O.D. (in.)	End of Life Rod O.D. (in.)	Rod I.D. (in.)	Avg. Tube Diameter (in.)	Wall Thickness (in.)	Hot Vol. Pressure at 300°C (MPa)	Cladding Stress (MPa)
8×8 SPC	0.484	0.4746	0.414	0.4443	0.0303	5.19	38.0
9×9 SPC w/ 2 water rods	0.424	0.4146	0.364	0.3893	0.0253	5.32	40.9
9×9 SPC w/ 1 water rod	0.423	0.4136	0.364	0.3888	0.0248	5.25	41.1
9×9 GE-11/13	0.44	0.4306	0.384	0.4073	0.0233	5.17	45.2
9×9 Atrium 9B SPC	0.433	0.4236	0.3808	0.4022	0.0214	5.32	50.0
10×10 SVEA-96	0.379	0.3696	0.3294	0.3495	0.0201	4.38	38.1
10×10 GE-12/14	0.404	0.3946	0.352	0.3733	0.0213	4.99	43.7
6x6 Dresden Thin Clad	*0.5625*	*0.5531*	*0.5105*	*0.5318*	*0.0213*	*5.77*	*72.5†*
7x7 Oyster Creek	*0.5700*	*0.5606*	*0.499*	*0.5298*	*0.0308*	*4.68*	*40.2*
8x8 Oyster Creek	*0.5015*	*0.4921*	*0.4295*	*0.4608*	*0.0313*	*4.78*	*35.2*
8x8 Quad† Westinghouse	*0.4576*	*0.4482*	*0.3996*	*0.4239*	*0.0243*	*6.33*	*55.2†*
8x8 TVA Browns Ferry	*0.483*	*0.4736*	*0.423*	*0.4483*	*0.0253*	*5.05*	*44.7*
9x9 SPC-5	*0.417*	*0.4076*	*0.364*	*0.3858*	*0.0218*	*5.38*	*47.6*

† These fuel types are evaluated separately for fuel cladding temperature limits.

Table 3.3.7

INITIAL PEAK CLADDING† TEMPERATURE LIMITS FOR TRANSPORT

Fuel Age (years)	Temperature Limits for PWR SNF (°C) [°F]	Temperature Limits for BWR†† SNF (°C) [°F]	Temperature Limits for 8x8 *and 6x6* Dresden-1, *Quad*+, and 6x6 Humboldt Bay SNF††† (°C) [°F]
5	382.3 [720]	398.2 [749]	*391.2 [736]* ~~396.0 [745]~~
6	370.2 [698]	382.3 [720]	*376.2 [709]* ~~380.7 [717]~~
7	347.0 [657]	357.9 [676]	*352.2 [666]* ~~356.2 [673]~~
10	341.6 [647]	351.4 [665]	*346.6 [656]* ~~349.9 [662]~~
15	334.1 [633]	344.9 [653]	*339.5 [643]* ~~343.4 [650]~~

† These limits are conservatively applied to stainless steel clad fuel assemblies, which actually have substantially higher limits.

†† 8x8 *and 6x6* GE Dresden-1, *Quad*†, and 6x6 GE Humboldt Bay SNF, for which cladding temperature limits are evaluated separately, are excluded from this group.

††† The 8x8 *and 6x6* GE Dresden-1, *Quad*†, and 6x6 GE Humboldt Bay fuel types are low heat emitting assemblies. The heat load ~~for these assembly types~~ ~~is~~ for Dresden fuel is limited to 115 watts per assembly (approximately 58% lower than the design basis maximum load for BWR fuel) (183.5 Watts Quad+ fuel). Consequently, these two assembly types are not deemed to be limiting.

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 MPC-24 fuel basket, 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,

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

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

C_o = *Assembly Geometry Factor*

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

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

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.

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]:

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

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 on the results of these conservative calculations, it is determined that the effects of this severe hypothetical condition do not exceed the abilities of the HI-STAR System.

3.4.1.1.18 HI-STAR Temperature Field With Low Heat Emitting Fuel

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. 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). 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. 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 579°F, which is substantially below the temperature limit for long-cooled fuel (~643°F).

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 (~14,400 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.

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 TN D-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).

---

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

‡ Allis-Chalmers stainless steel clad fuel assemblies.

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)	1.108	1.495	1.954
MPC-68 (Zircaloy Clad Fuel)	0.959	1.188	1.432
MPC-24 (Stainless Steel Clad Fuel)	0.995	1.321	1.700[a]
MPC-68 (Stainless Steel Clad Fuel)	0.931	1.125	1.311[b]
MPC-68 (Dresden-1 8x8 in canister)	*0.861*	*1.055*	*1.242*

(a) 13% lower effective thermal conductivity than corresponding zircaloy-fueled basket
(b) 9% lower effective thermal conductivity than corresponding zircaloy-fueled basket

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.

Table 4.1.1

SUMMARY OF CONTAINMENT BOUNDARY DESIGN SPECIFICATIONS

Design Attribute	Design Rating	
	Primary (Overpack) 10CFR71.51	Secondary (MPC-68F) 10CFR71.63(b)
Closure Plate Mechanical Seals:†† Design Temperature Pressure Rating Design Leakage Rate	 1200°F ~~10,000~~*1,000* psig 1X10^{-6} cm^3/s, Helium	N/A
Overpack Vent and Drain Port Cover Plate Mechanical Seals: †,†† Design Temperature Pressure Rating Design Leakage Rate	 1300°F ~~5000~~*1,000* psig 1×10^{-6} cm^3/sec, Helium	N/A
Overpack Vent and Drain Port Plug Mechanical Seals:†† Design Temperature Pressure Rating Design Leakage Rate	 1300°F ~~5,000~~*1,000* psig 1×10^{-6} cm^3/sec, Helium	N/A
Standard Leakage Rate	4.3 x 10^{-6} std cm^3/s, He	5.0 x 10^{-6} std cm^3/s, Helium
Sensitivity-Standard Leakage Rate	2.15 x 10^{-6} std cm^3/s, He	2.5 x 10^{-6} std cm^3/s, Helium

† No credit is taken for the overpack vent and drain port cover plate seals as part of the containment boundary. Specifications are provided for information.

†† Per manufacturer's recommended operating limits.

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 was conservatively based on the B&W 15x15 fuel assembly with a burnup of 40,000 MWD/MTU, 5 years of cooling time, and an enrichment of 3.4%. The inventory for the MPC-68 was based the GE 7x7 fuel assembly with a burnup of 40,000 MWD/MTU, 5 years of cooling time, and 3.0% 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. Table 4.2.2 presents the isotope inventory used in the calculation.

A. Source Activity Due to Crud Spallation from Fuel Rods

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 and MPC-68; 18 years for the MPC-68F) using the basic radioactive decay equation:

$$A(t) = A_o e^{-\lambda t} \quad (4\text{-}1)$$

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 and MPC-68; 18 years for the MPC-68F)

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

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

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

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

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

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

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:

$$C_{crud} = \frac{f_C\ M_A\ N_A}{V} \quad (4\text{-}2)$$

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=3x10^{-5}).

The calculation for normal transport conditions of 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 64 assemblies in the MPC-68F were assumed to have a 3% cladding rupture. Therefore, f_B for an MPC-68F containing fuel debris is:

$$
\begin{aligned}
f_B &= (0.03)\frac{64}{68} + (1.0)\frac{4}{68} \\
f_B &= 0.087
\end{aligned}
\qquad \textbf{(4-3)}
$$

The activity concentration inside the containment vessel due to fines being released from cladding breaches is given by:

$$
C_{fines} = \frac{f_f \, I_{fines} \, N_A \, f_B}{V} \qquad \textbf{(4-4)}
$$

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:

$$C_{gases} = \frac{f_g \, I_{gases} \, N_A \, f_B}{V} \qquad \textbf{(4-5)}$$

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] [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:

$$C_{vol} = \frac{f_v \, I_{vol} \, N_A \, f_B}{V} \tag{4-6}$$

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 [^{90}Sr, ^{137}Cs, ^{134}Cs, ^{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,

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 (~~1.90x10^-5~~ *1.93x10^-5* cm^3/s under normal conditions of transport) was conservatively selected and used to determine the standard leakage rate using the ratio presented in Equation 4-13.

$$L_{@P_a} = L_{@P_u} \frac{P_u}{P_a} \quad \textbf{(4-13)}$$

where:

$L_{@Pa}$	is the allowable leakage rate at the average pressure [cm^3/s]
$L_{@Pu}$	is the allowable leakage rate at the upstream pressure [cm^3/s]
P_u	is the upstream pressure [ATM],
P_a	is the average pressure; $P_a = (P_u + P_d)/2$ [ATM], and
P_d	is the downstream pressure [ATM].

Substituting ~~1.90x10^-5~~ *1.93x10^-5* cm^3/s for $L_{@Pu}$, 7.8 ATM (the upstream pressure reported in Table 4.2.12) for P_u, and 4.4 ATM ($(P_u + P_d)/2$ where P_u and P_d are presented in Table 4.2.12) the allowable leakage rate at the average pressure is determined. The corresponding leakage rate at the average pressure was ~~3.37x10^-5~~ *3.41x10^-5* cm^3/s.

4.2.5.8 Standard Leakage Rates

The leakage rate discussed thus far was determined at operating conditions (see normal conditions in Table 4.2.12). The following provides details of the methodology used to convert the allowable leakage rate (~~3.37x10^-5~~ *3.41x10^-5* cm^3/s) to a standard leakage rate at 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 4-14 for D, the leak hole diameter. The capillary length required for Equation 4-14 for the primary containment was conservatively chosen as the closure plate inner seal seating width which is 0.25 cm; for the secondary containment (MPC-68F), the capillary length was conservatively chosen to be the MPC lid closure weld thickness which is 1.25 inches thick (3.175 cm).

$$L_{@P_a} = \left[\frac{2.49 \times 10^6 \, D^4}{a \, u} + \frac{3.81 \times 10^3 \, D^3 \sqrt{\frac{T}{M}}}{a \, P_a}\right] [P_u - P_d] \qquad \textbf{(4-14)}$$

where:

$L_{@P_a}$ is the allowable leakage rate at the average 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 leak hole diameter was determined by solving Equation 4-14 for 'D' where $L_{@P_a}$ is equal to ~~3.37×10^{-5}~~ *3.41×10^{-5}* cm^3/s and using the parameters for normal conditions of transport presented in Table 4.2.12. The corresponding leak hole diameter was determined to be ~~3.28×10^{-4}~~ *3.29×10^{-4}* cm..

Using this leak hole diameter (~~3.28×10^{-4}~~ *3.29×10^{-4}* cm.), and the temperature and pressures for test conditions provided in Table 4.1.12, Equation 4-14 was solved for the standard leakage rate (~~6.44×10^{-6}~~ *8.20×10^{-6}* std cm^3/s, helium) at test conditions. For additional conservatism to ensure compliance with 10CFR71.51, this standard leakage rate (~~6.44×10^{-6}~~ *8.20×10^{-6}* std cm^3/s, helium) was then 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 standard leakage rate and sensitivity were determined following the basic methodology described above. To determine this leakage rate, only the plutonium inventory for the GE 6x6 MOX fuel assembly was used as this inventory bounds the standard GE 6x6 fuel assembly Plutonium inventory. Table 4.2.11 contains the plutonium inventory for the MOX fuel used in this evaluation.

As discussed in 4.2.5.2, Equation 4-3 presents the methodology to determine f_B for an MPC-68F containing fuel debris. This f_B was applied in determining the source activity due to Plutonium. The calculation for normal transport conditions of 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 64 assemblies in the MPC-68F were assumed to have a 3% cladding rupture. Therefore, f_B for an MPC-68F containing fuel debris under normal conditions of transport is 0.087. 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., $f_{Pu}=3\times10^{-5}$). 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/cm³],
f_{Pu}	is the fraction of a fuel rod's mass released as Plutonium ($f_f = 3\times10^{-5}$),
I_{Pu}	is the total Plutonium inventory of one GE 6x6 MOX assembly [Ci/assy],
N_A	is the number of GE 6x6 MOX assemblies (68),
f_B	is the fraction of rods that develop cladding breaches (f_B=0.087 for normal conditions of transport and f_B=1.0 for accident conditions), and
V	is the free volume inside the containment vessel [cm³] 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 (0.0297 Ci). 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 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 (3.77×10^{-5} cm^3/s under normal conditions of transport) was conservatively selected and used to determine the standard leakage rate using the ratio presented in Equation 4-13. The corresponding leakage rate at average pressure was 6.68×10^{-5} cm^3/s.

As discussed in 4.2.5.8, the leakage rate at average pressure (6.68×10^{-5} cm^3/s) was then converted to a standard leakage rate at test conditions using the equations for molecular and continuum flow provided in NUREG/CR-6487 (Equation 4-14). The capillary length required for Equation 4-14 for the secondary containment (MPC-68F) was conservatively chosen to be the MPC lid closure weld thickness which is assumed to be 1.25 inches thick (3.175 cm). Equation 4-14 was solved for D, the leak hole diameter ($D=7.73 \times 10^{-4}$ cm) and then using this leak hole diameter, and the temperature and pressures for test conditions (Table 4.1.12), Equation 4-14 was solved for the standard leakage rate at test conditions (1.88×10^{-5} std cm^3/s, helium). For additional conservatism to ensure compliance with 10CFR71.63(b), this leakage rate (1.88×10^{-5} std cm^3/s, helium) was conservatively reduced and is presented in Table 4.1.1.

4.2.5.10 Leak Test Sensitivity

The sensitivity for the leakage test procedures is equal to one half of the allowable standard leakage rate. The HI-STAR 100 containment packaging tests in Chapter 8 incorporate the appropriate leakage test procedure sensitivity. The standard leakage rates for the HI-STAR 100 containment packaging with their corresponding sensitivity are presented in Table 4.1.1.

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	~~28.6~~ *29.2*
MPC-68	~~17.9~~ *18.2*
MPC-68F	14.4
Accident Conditions	
MPC-24	~~32.5~~ *33.2*
MPC-68	~~26.8~~ *27.3*
MPC-68F	14.9

Table 4.2.7

RADIONUCLIDE RELEASE RATES

	Effective A_2 (Ci)	Allowable Release Rate (R_N or R_A) (Ci/s)
Normal Transport Conditions		
MPC-24	28.6	~~7.96E-09~~ *8.12E-09*
MPC-68	17.9	~~4.98E-09~~ *5.06E-09*
MPC-68F Primary Containment	14.5	4.02E-09
Accident Conditions		
MPC-24	32.5	~~5.36E-05~~ *5.47E-05*
MPC-68	26.8	~~4.42E-05~~ *4.51E-05*
MPC-68F Primary Containment	14.9	2.46E-05

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	1.43E-04	~~5.56E-05~~ *5.67E-05*
MPC-68	2.63E-04	~~1.90E-05~~ *1.93E-05*
MPC-68F Primary Containment	9.56E-05	4.20E-05
Accident Conditions		
MPC-24	4.46E-03	~~1.20E-02~~ *1.23E-02*
MPC-68	5.88E-03	~~7.51E-03~~ *7.67E-03*
MPC-68F Primary Containment	1.00E-03	2.45E-02

Table 4.2.12

PARAMETERS FOR NORMAL, HYPOTHETICAL ACCIDENT AND STANDARD CONDITIONS

Parameter	Normal (helium)	Hypothetical Accident (helium)	Standard (Test Conditions, helium)
P_u	114.7 psia (7.8 ATM)	139.7 psia (9.5 ATM)	Primary: 1.68 ATM
			Secondary: 2.0 ATM
P_d	14.7 psia (1 ATM)	14.7 psia (1 ATM)	14.7 psia (1 ATM)
T	400°C (673 K)	1058°F (843 K)	373 K
M	4 g/mol	4 g/mol	4 g/mol
u	0.0341 cP	0.0397 cP	0.0231 cP
a	Primary: 0.25 cm	Primary: 0.25 cm	Primary: 0.25 cm
	Secondary: *3.175* ~~1.9~~ cm	Secondary: *3.175* ~~1.9~~ cm	Secondary: *3.175* ~~1.9~~ cm

CHAPTER 5: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STAR 100 System is presented in this chapter. The HI-STAR 100 System is designed to accommodate different MPCs within one standard HI-STAR 100 overpack. The MPCs are designated as MPC-24 (24 PWR fuel assemblies) and MPC-68 (68 BWR fuel assemblies).

In addition to housing intact PWR and BWR fuel assemblies, the HI-STAR 100 System is designed to transport damaged BWR fuel assemblies and BWR fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 1.2.3. Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. 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 the MPC-68F. Only the fuel assemblies in the Dresden 1 and Humboldt Bay fuel assembly classes identified in Table 1.2.9 are authorized as contents for transport in the HI-STAR 100 system as either damaged fuel or fuel debris.

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

This chapter contains the following information:

- A description of the shielding features of the HI-STAR 100 System.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STAR 100 System.
- Analyses for each of the HI-STAR 100 Systems content conditions to show that the 10CFR71.47 radiation limits are met during normal conditions of transport and that the 10CFR71.51 dose rate limit is not exceeded following hypothetical accident conditions.
- Analyses which demonstrate that the storage of damaged fuel in the HI-STAR 100 System is bounded by the BWR intact fuel analysis during normal and hypothetical accident conditions.

5.1 DISCUSSION AND RESULTS

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

- Gamma radiation originating from the following sources

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

- Neutron radiation originating from the following sources

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

Shielding from gamma radiation is provided by the steel structure of the MPC and overpack. In order for the neutron shielding to be effective, the neutrons must be thermalized and then absorbed in a material of high neutron cross section. In the HI-STAR 100 System design, a neutron shielding material, Holtite-A, is used to thermalize the neutrons. Boron carbide, dispersed in the neutron shield, utilizes the high neutron absorption cross section of ^{10}B to absorb the thermalized neutrons.

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

The design basis intact zircaloy clad fuels used in calculating the dose rates presented in this chapter are the B&W 15x15 (with zircaloy and non-zircaloy incore spacers) and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6, damaged, and mixed oxide (MOX) fuel assemblies are the GE 6x6. Table 1.2.13 specifies the acceptable intact zircaloy clad fuel characteristics for transport. Table 1.2.14 specifies the acceptable damaged and MOX zircaloy clad fuel characteristics for transport.

The design bases intact stainless steel clad fuels are the WE 15x15 and the AC 10x10, for PWR and BWR fuel types, respectively. Table 1.2.19 specifies the acceptable fuel characteristics of stainless steel clad fuel for transport.

Table 1.2.20 specifies, in tabular form, the minimum enrichment, burnup and cooling time combinations for spent nuclear fuel that were analyzed for transport in the MPC-24 and MPC-68. Each combination provides a dose rate equal to or below the maximum values reported in this section. This table represents the fuel assembly acceptance criteria.

Table 1.2.20 was developed in a two stage process. First, the burnup and cooling time combinations that produced assembly decay heat rates equal to the thermal limits specified in Figure 1.2.12 were calculated. Second, the dose rates at the various locations were calculated for these burnup and cooling time combinations and compared to the regulatory limits. In some cases, the burnup, for a specified cooling time, had to be reduced to meet the dose rate limits. Therefore, Table 1.2.20 is based on both the maximum permissible decay heat per assembly and the regulatory dose rate limits. The burnup and cooling time combinations analyzed in this chapter are equivalent to or bound the acceptable burnup and cooling time combinations in Table 1.2.20. The dose rates from the burnup and cooling time combination which provided the highest dose rates at the midplane of the cask for each location (surface and 2 meter - normal condition, and 1 meter - hypothetical accident condition) are reported in this section. As a result, the burnup and cooling time combinations reported in this section may be different between locations. Dose rates for each combination calculated are listed in Section 5.4.

Unless otherwise stated, all dose rates reported in this chapter are average surface dose rates. The effect of radiation peaking due to azimuthal variations in the fuel loading pattern and the steel radial channels is specifically addressed in Subsection 5.4.1.

5.1.1 Normal Operations

The 10CFR71.47 external radiation requirements during normal transport operations for an exclusive use shipment are:

1. 200 mrem/hr (2 mSv/hr) on the external surface of the package.

2. 200 mrem/hr (2 mSv/hr) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle.

3. 10 mrem/hr (0.1 mSv/hr) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle).

4. 2 mrem/h (0.02 mSv/hr) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10CFR20.1502.

The external surface of the HI-STAR 100 System during normal transportation is defined as the outer surface of the impact limiters and the outer radial surface of the overpack in the region between the impact limiters.

The HI-STAR 100 System will be transported on either a flat-bed rail car, heavy haul vehicle, or a barge. The smallest width of a transport vehicle is equivalent to the width of the impact limiters. Therefore, the vertical planes projected by the outer side edges of the transport vehicle are equivalent to the outer edge of the impact limiters. The minimum length of any transport vehicle will be 12 feet longer than the length of the overpack, with impact limiters attached. The HI-STAR 100 System will be conservatively positioned a minimum of 6 feet from either end of the transport vehicle. Therefore, the vertical planes projected from the outer edge of the ends of the vehicle will be taken as the end of the top impact limiter and 6 feet from the end of the bottom impact limiter.

Figure 5.1.1 shows the HI-STAR 100 System during normal transport conditions. The impact limiters are outlined on the figure and various dose point locations are shown on the surface of the HI-STAR 100 System. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width. Each of the dose locations on the surface of the HI-STAR 100 System has a corresponding location at 2 meters from the surface of the transport vehicle as defined above.

Tables 5.1.1 through 5.1.3 provide the maximum dose rates on the surface of the system during normal transport conditions for the MPC-24 and MPC-68 with design basis intact zircaloy clad fuel. Tables 5.1.4 through 5.1.6 list the maximum dose rates two meters from the edge of the transport vehicle during normal conditions. Section 5.4 provides a detailed list of dose rates at several cask locations for all burnup and cooling times analyzed.

Subsections 5.2.1 and 5.2.2 list the gamma and neutron sources for the design basis zircaloy clad intact, zircaloy clad damaged and MOX fuel assemblies. Since the source strengths of the damaged and MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the damaged and MOX fuel dose rates for normal conditions are bounded by the MPC-68 analysis with design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either damaged or MOX fuel for normal conditions is required to demonstrate that the MPC-68 with damaged fuel or MOX fuel will meet the normal condition regulatory requirements.

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

Subsection 5.4.5 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Subsection 5.2.3 lists the gamma and neutron sources for the design basis intact stainless steel clad fuels. The dose rates from these fuels are provided in Subsection 5.4.4.

Tables 5.1.4 through 5.1.6 show that the dose rate at Dose Location #5 (the top of the HI-STAR 100 System, see Figure 5.1.1) at 2 meters from the edge of the transport vehicle is less than 2 mrem/hr. It is, therefore, recommended that the HI-STAR 100 System be positioned such that the top impact limiter is facing the normally occupied space. If this is the orientation, radiation dosimetry will not be required as long as the normally occupied space is a minimum of 2 meters from the impact limiter on the top of the HI-STAR 100 System. If a different orientation is chosen for the HI-STAR 100 System, the dose rate in the normally occupied space will have to be evaluated against the dose requirement for the normally occupied space to determine if radiation dosimetry is required.

The analyses summarized in this section demonstrate the HI-STAR 100 System's compliance with the 10CFR71.47 limits.

5.1.2 Hypothetical Accident Conditions

The 10CFR71.51 external radiation dose limit for design basis accidents is:

- The *external* radiation dose *rate* shall not exceed 1 rem/hr (10 mSv/hr) at 1 m (40 in.) from the external surface of the package.

The hypothetical accident conditions of transport have two bounding consequences which affect the shielding materials. They are the damage to the neutron shield as a result of the design basis fire and damage of the impact limiters as a result of the 30 foot drop. In a conservative fashion, the dose analysis assumes that as a result of the fire, the neutron shield is completely destroyed and replaced by a void. Additionally, the impact limiters are assumed to have been lost. These are highly conservative assumptions since some portion of the neutron shield would be expected to remain after the fire as the neutron shield material is fire retardant, and the impact limiters have been shown by 1/4-scale testing to remain attached following impact (see Appendix 2.H).

Throughout the hypothetical accident condition the axial location of the fuel will remain fixed within the MPC because of the fuel spacers or by the MPC lid and baseplate if spacers are not used. Chapter 2 provides an analysis to show that the fuel spacers do not fail under all normal and hypothetical accident conditions. Chapter 2 also shows that the inner shell, intermediate shell, radial channels, and outer enclosure shell of the overpack remain unaltered throughout the hypothetical accident conditions. Localized damage of the overpack outer enclosure shell could be experienced during the pin puncture. However, the localized deformations will have only a negligible impact on the dose rate at 1 meter from the surface.

Figure 5.1.2 shows the HI-STAR 100 System after the postulated accident. The various dose

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

5.2.1 Gamma Source

Tables 5.2.3 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design bases intact fuels for the MPC-24, MPC-68, and the design basis damaged fuel. Table 5.2.16 provides the gamma source in MeV/s and photons/s for the design basis MOX fuel. NUREG-1617 [5.2.1] states that "In general, only gammas from approximately 0.8 MeV-2.5 MeV will contribute significantly to the external radiation levels." However, specific analysis for the HI-STAR 100 system has revealed that, due to the magnitude of the gamma source in the energy range just below 0.8 MeV, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Photons with energies below 0.45 MeV are too weak to penetrate the steel of the overpack, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose. As discussed earlier, the MPC-24 and the MPC-68 are analyzed for transportation of spent nuclear fuel with varying minimum enrichments, burnup levels and cooling times. This section provides the radiation source for each of the burnup levels and cooling times evaluated.

The primary source of activity in the non-fuel regions of an assembly arise from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel element is the steel and inconel structural material. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.3] indicates that the ^{59}Co impurity level in steel is 800 ppm or 0.8 gm/kg and in inconel is approximately 4700 ppm or 4.7 gm/kg. In the early to mid 1980s, the fuel vendors reduced the ^{59}Co impurity level in both inconel and steel to less than 500 ppm or 0.5 gm/kg. Prior to that, the impurity level in inconel in fuel assemblies was typically less than 1200 ppm or 1.2 gm/kg. Nevertheless, a conservative ^{59}Co impurity level of 1.0 gm/kg was used for the stainless steel end fittings and a highly conservative impurity level of 4.7 gm/kg was used for the inconel.

PWR fuel assemblies are currently manufactured with zircaloy incore grid spacers (the plenum spacer and the lower spacer are still inconel in some cases). However, earlier assemblies were manufactured with inconel incore grid spacers. Since the mass of the spacers is significant and since the cobalt impurity level assumed for inconel is very conservative, the Cobalt-60 activity from the incore spacers contributes significantly to the external dose rate. As a result, separate burnup and cooling times were developed for PWR assemblies that utilize zircaloy and non-zircaloy incore spacers. Since steel has a lower cobalt impurity level than inconel, any zircaloy clad PWR assemblies with stainless steel grid spacers are bounded by the analysis performed in this chapter utilizing inconel grid spacers. The BWR assembly grid spacers are zircaloy, however, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 was taken from References [5.2.3~~2~~], [5.2.4], and [5.2.5]. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. These masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ^{59}Co impurity level, results in a conservative estimate of the ^{60}Co activity.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.2~~3~~] and is described here.

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

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.7. These scaling factors were taken from Reference [5.2.2~~3~~].

Tables 5.2.8 through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations in the non-fuel regions of the assemblies for the MPC-24 and MPC-68. The design basis damaged and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR intact design basis fuel. This is a conservative assumption as the design basis damaged fuel and MOX fuel are limited to a significantly lower burnup and longer cooling time than the intact design basis zircaloy clad fuel.

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

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel

22,500 MWD/MTU and 15-year cooling as specified in Table 1.2.19. This assembly type is discussed further in Subsection 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher UO_2 mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis damaged fuel assembly and the design basis intact 6x6 fuel assembly are identical, the analysis presented in Subsection 5.4.2 for the damaged fuel assembly also demonstrates the acceptability of transporting intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

5.2.5.3 Decay Heat Loads

The decay heat values per assembly were calculated using the methodology described in Section 5.2. The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time limits. As demonstrated in Tables 5.2.26 and 5.2.27, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, Tables 1.2.10 and 1.2.11 limit the heavy metal mass *of the design basis fuel assembly classes* to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.

As further demonstration that the decay heat values (calculated using the design basis fuel assemblies) are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.28. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

The heavy metal mass of the non-design basis fuel assembly classes in Tables 1.2.10 and 1.2.11 are limited to the masses used in Tables 5.2.24 and 5.2.25. No margin is applied between the allowable mass and the analyzed mass of heavy metal for the non-design basis fuel assemblies. This is acceptable because additional assurance that the decay heat values for the non-design basis fuel assemblies are bounding values is obtained by using the decay heat values for the design basis fuel assemblies in determining the acceptable loading criteria for all fuel *assemblies. As mentioned above, Table 5.2.28 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.*

5.2.6 Thoria Rod Canister

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

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.29 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.30 and 5.2.31 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.6 and 5.2.14 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Subsection 5.4.6 provides a further discussion of the thoria rod cansiter and its acceptablity for transport in the HI-STAR 100 System.

5.2.7 Fuel Assembly Neutron Sources

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

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Subsection 5.4.5 to demonstrate that they are acceptable for transport in the HI-STAR 100 System. Currently these are the only neutron source permitted for transport in the HI-STAR 100 System.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL

	PWR	BWR
No. of Water Rods/Guide Tubes	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.0160	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.55572 (inconel) 0.27252 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)†	0.33 (inconel springs)

~~Notes:~~
~~The mass of inconel used for the PWR incore grid spacers~~

† This mass of inconel was used for fuel assemblies with non-zircaloy grid spacers. For fuel assemblies with zircaloy grid spacers the mass was 0.0. However, the mass of the inconel and steel in the other assembly components are identical for assemblies with zircaloy and non-zircaloy incore grid spacers.

Table 5.2.29
DESCRIPTION OF FUEL ASSEMBLY USED TO ANNALYZE THORIA RODS IN THE THORIA ROD CANISTER

	BWR
Fuel type	*8x8*
Active fuel length (in.)	*110.5*
No. of UO_2 fuel rods	*55*
No. of UO_2/ThO_2 fuel rods	*9*
Rod pitch (in.)	*0.523*
Cladding material	*zircaloy*
Rod diameter (in.)	*0.412*
Cladding thickness (in.)	*0.025*
Pellet diameter (in.)	*0.358*
Pellet material	*98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods*
Pellet density (gm/cc)	*10.412*
Enrichment (w/o ^{235}U)	*93.5 in UO_2 for UO_2/ThO_2 rods and 1.8 for UO_2 rods*
Burnup (MWD/MTIHM)	*16,000*
Cooling Time (years)	*18*
Specific power (MW/MTIHM)	*16.5*
Weight of THO_2 and UO_2 (kg)†	*121.46*
Weight of U (kg)†	*92.29*
Weight of Th (kg)†	*14.74*

† *Derived from parameters in this table.*

Table 5.2.30

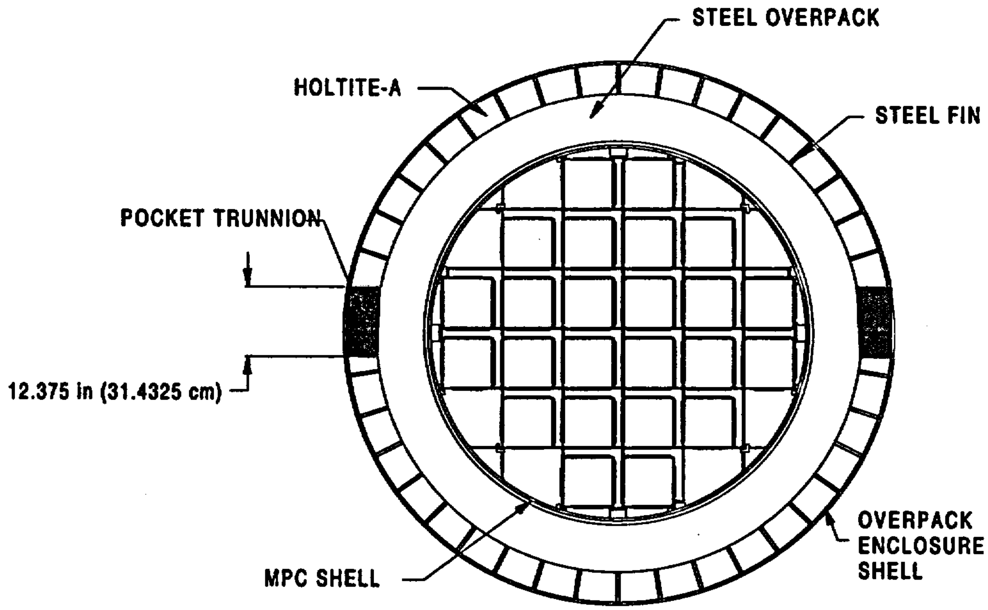
CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

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

Table 5.2.31

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy (MeV)	*Upper Energy (MeV)*	*16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)*
1.0e-01	4.0e-01	5.65e+2
4.0e-01	9.0e-01	3.19e+3
9.0e-01	1.4	6.79e+3
1.4	1.85	1.05e+4
1.85	3.0	3.68e+4
3.0	6.43	1.41e+4
6.43	20.0	1.60e+2
Totals		7.21e+4



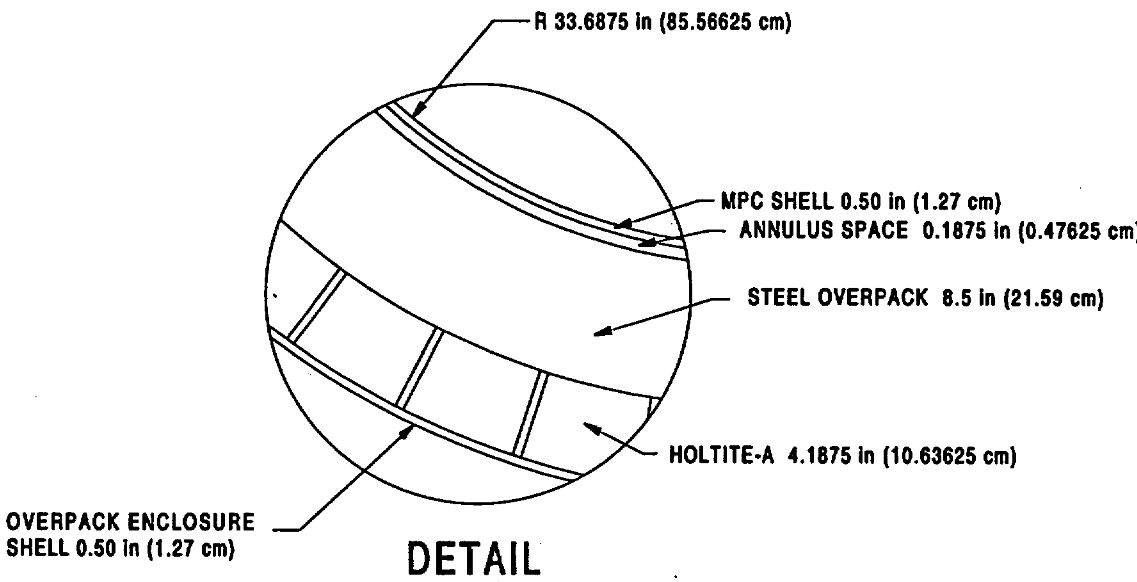


FIGURE 5.3.9; HI-STAR 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW SHOWING THE THICKNESS OF THE MPC ~~SHIELD~~ *SHELL* AND OVERPACK AS MODELED IN MCNP

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.

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	6.38E+5	N/A	Table 5.2.13 39.5 GWD/MTU and 15 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 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	0.9383
14x14B	4.6	0.9323
14x14C	4.6	0.9~~361~~*400*
14x14D	4.0	0.8576
15x15A	4.1	0.9301
15x15B	4.1	0.9473
15x15C	4.1	0.9444
15x15D	4.1	0.9440
15x15E	4.1	0.9475
15x15F	4.1	0.9478 ††
15x15G	4.0	0.8986
15x15H	*3.8*	*0.9411*
16x16A	4.6	0.9383
17x17A	4.0	0.9452
17x17B	4.0	0.9436
17x17C	4.0	0.9427

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 0.9466.

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††	0.76~~02~~*888*†††
6x6B‡	2.7††	0.76~~11~~*824*†††
6x6C	2.7††	0.8021†††
7x7A	2.7††	0.797~~3~~*4*†††
7x7B	4.2	0.93~~7~~*886*
8x8A	2.7††	0.7697~~85~~†††
8x8B	4.2	0.9~~368~~*416*
8x8C	4.2	0.9425
8x8D	4.2	0.9~~366~~*403*
8x8E	4.2	0.9312
8x8F	*4.2*	*0.9140*

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.

†† 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	0.9*422*~~388~~
9x9C	4.2	0.9395
9x9D	4.2	0.939~~2~~*4*
9x9E	4.~~2~~*1*	0.94~~06~~*24*
9x9F	4.~~2~~*1*	0.9~~377~~*424*
10x10A	4.2	0.9457††
10x10B	4.2	0.9436
10x10C	4.2	0.~~8990~~*9021*
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††	0.7~~602~~888
6x6B†††	2.7	0.7~~611~~824
6x6C	2.7	0.8021
7x7A	2.7	0.797~~3~~4
8x8A	2.7	0.7685

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.

assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, the authorized contents (Chapter 1) are defined in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R). The results of this study are shown in Table 6.2.3, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod thickness, (5) artificially replacing the Zircaloy water rod tubes with water, and (6) maximizing the channel thickness. These results, and the many that follow, justify the approach for using bounding dimensions for defining the authorized contents. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of authorized contents.

As mentioned, the bounding approach used in these analyses often results in a maximum k_{eff} value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

6.2.2 PWR Fuel Assemblies in the MPC-24

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum k_{eff} of 0.9478). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.16) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.4 through 6.2.189 for the fully flooded condition.

Tables 6.2.4 through 6.2.189 show the maximum k_{eff} values for the assembly classes that are acceptable for storage in the MPC-24. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations for the MPC-24 were performed for a ^{10}B loading of 0.020 g/cm^2, which is 75% of the minimum loading, 0.0267 g/cm^2, specified on BM-1478, Bill of Materials for 24-Assembly HI-STAR 100 PWR MPC, in Section 1.4. The maximum allowable enrichment in the MPC-24 varies from 4.0 to 4.6 wt% ^{235}U, depending on the assembly class, and is defined in Tables 6.2.4 through 6.2.189. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.4 through 6.2.1~~8~~*9* are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.4). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.7), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

6.2.3 BWR Fuel Assemblies in the MPC-68

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum k_{eff} of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 are summarized in Tables 6.2.~~19~~*20* through 6.2.3~~4~~*6* for the fully flooded condition. In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.~~19~~*20* through 6.2.3~~4~~*6* show the maximum k_{eff} values for assembly classes that are acceptable for storage in the MPC-68. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 were performed with a ^{10}B loading of 0.0279 g/cm^2, which is 75% of the minimum loading, 0.0372 g/cm^2, specified on BM-1479, Bill of Materials for 68-Assembly HI-STAR 100 BWR MPC, in Section 1.4. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a ^{10}B loading of 0.0067 g/cm^2. The maximum allowable enrichment in the MPC-68 varies from 2.7 to 4.2 wt% ^{235}U, depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.2 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68.

Tables 6.2.~~19~~*20* through 6.2.34*6* are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.2~~3~~*4*), the authorized contents dimensions are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain assemblies with thicker channels (e.g., 120 mils), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.~~19~~*20*). All of the maximum k_{eff} values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum k_{eff} values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the specification of authorized contents has no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of authorized contents. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum k_{eff} value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils thick; and (3) the active fuel length of the full length rods is 155 inches. Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

6.2.4 Damaged BWR Fuel Assemblies and BWR Fuel Debris

In addition to storing intact PWR and BWR fuel assemblies, the HI-STAR 100 System is designed to store damaged BWR fuel assemblies and BWR fuel debris. Damaged fuel assemblies and fuel debris are defined in Chapter 1. Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. *Two different DFC types with slightly different cross sections are considered.* 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. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4 *for both DFC types.*

Tables 6.2.3~~5~~*7* through 6.2.~~39~~*41* show the maximum k_{eff} values for the six assembly classes that may be stored as damaged fuel or fuel debris. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum loading, 0.0089 g/cm^2. However, because the practical manufacturing lower limit for minimum ^{10}B loading is 0.01 g/cm^2, the minimum ^{10}B loading of 0.01 g/cm^2 is specified on BM-1479, Bill of Materials for 68-Assembly HI-STAR 100 BWR MPC, in Section 1.4, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt% ^{235}U, while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt% ^{235}U in the specification of authorized contents. Therefore, the maximum k_{eff} values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.3~~5~~*7* through 6.2.~~39~~*41* for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt% ^{235}U enrichment has the highest reactivity (maximum k_{eff} of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual ^{10}B loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a minimum ^{10}B loading of 0.0089 g/cm^2, which conservatively bounds damaged BWR fuel assemblies in a standard MPC-68 with a minimum ^{10}B loading of 0.0372 g/cm^2, damaged BWR fuel assemblies may also be stored in the standard MPC-68. However, fuel debris is limited to the MPC-68F by the specification of authorized contents in Chapter 1.

Tables 6.2.3~~5~~*7* through 6.2.~~39~~*41* are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected to define the authorized contents and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.3~~7~~*9*), the authorized contents dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

6.2.5 Thoria Rod Canister

Additionally, th HI-STAR 100 System is designed to store a Thoria Rod Canister in the MPC68 or MPC68F. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC68 or MPC68F is very low compared to the reactivity of the approved fuel assemblies (The ^{235}U content of these rods *corresponds to UO_2 rods with an initial enrichment of approximately 1.7 wt% ^{235}U). It is therefore permissible to store the Thoria Rod Canister together with any other approved content in a MPC68 or MPC68F. Specifications of the canister and the Thoria Rods that are used in the criticality evaluation are given in Table 6.2.42. The criticality evaluation is presented in Subsection 6.4.6.*

Table 6.2.1 (page 1 of 46)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694	36	0.5645	0.0350	0.4940	110.0	0	n/a	n/a	0.060	4.290
6x6A02	Zr	0.694	36	0.5645	0.0360	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A03	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A04	Zr	0.694	36	0.5550	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A05	*Zr*	*0.696*	*36*	*0.5625*	*0.0350*	*0.4820*	*110.0*	*0*	*n/a*	*n/a*	*0.060*	*4.290*
6x6A06	*Zr*	*0.696*	*35*	*0.5625*	*0.0350*	*0.4820*	*110.0*	*1*	*0.0*	*0.0*	*0.060*	*4.290*
6x6A07	*Zr*	*0.700*	*36*	*0.5555*	*0.03525*	*0.4780*	*110.0*	*0*	*n/a*	*n/a*	*0.060*	*4.290*
6x6A08	*Zr*	*0.710*	*36*	*0.5625*	*0.0260*	*0.4980*	*110.0*	*0*	*n/a*	*n/a*	*0.060*	*4.290*
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B02	Zr	0.694	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B03	*Zr*	*0.696*	*36*	*0.5625*	*0.0350*	*0.4820*	*110.0*	*0*	*n/a*	*n/a*	*0.060*	*4.290*
6x6B04	*Zr*	*0.696*	*35*	*0.5625*	*0.0350*	*0.4820*	*110.0*	*1*	*0.0*	*0.0*	*0.060*	*4.290*
6x6B05	*Zr*	*0.710*	*35*	*0.5625*	*0.0350*	*0.4820*	*110.0*	*1*	*0.0*	*0.0*	*0.060*	*4.290*
6x6C Assembly Class												
6x6C01	Zr	0.740	36	0.5630	0.0320	0.4880	77.5	0	n/a	n/a	0.060	4.542
7x7A Assembly Class												
7x7A01	Zr	0.631	49	0.4860	0.03~~30~~*28*	0.4110	~~79~~*80*	0	n/a	n/a	0.060	4.542

Table 6.2.1 (page 2 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738	49	0.5630	0.0320	0.4870	150	0	n/a	n/a	0.080	5.278
7x7B02	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.102	5.291
7x7B03	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.080	5.278
7x7B04	Zr	0.738	49	0.5700	0.0355	0.4880	150	0	n/a	n/a	0.080	5.278
7x7B05	Zr	0.738	49	0.5630	0.0340	0.4775	150	0	n/a	n/a	0.080	5.278
7x7B06	*Zr*	*0.738*	*49*	*0.5700*	*0.0355*	*0.4910*	*150*	*0*	*n/a*	*n/a*	*0.080*	*5.278*
8x8A Assembly Class												
8x8A01	Zr	0.523	64	0.4120	0.0250	0.3580	110	0	n/a	n/a	0.100	4.290
8x8A02	*Zr*	*0.523*	*63*	*0.4120*	*0.0250*	*0.3580*	*120*	*0*	*n/a*	*n/a*	*0.100*	*4.290*

Table 6.2.1 (page ~~2~~*3* of 4*6*)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B02	Zr	0.636	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B03	Zr	0.640	63	0.4930	0.0340	0.4160	150	1	0.493	0.425	0.100	5.278
8x8B04	*Zr*	*0.642*	*64*	*0.5015*	*0.0360*	*0.4195*	*150*	*0*	*n/a*	*n/a*	*0.100*	*5.278*
8x8C Assembly Class												
8x8C01	Zr	0.641	62	0.4840	0.0350	0.4050	150	2	0.484	0.414	0.100	5.278
8x8C02	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.000	no channel
8x8C03	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.080	5.278
8x8C04	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C05	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C06	Zr	0.640	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.100	5.278
8x8C07	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C08	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.493	0.425	0.100	5.278
8x8C09	Zr	0.640	62	0.4930	0.0340	0.4160	150	2	0.493	0.425	0.100	5.278
8x8C10	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C11	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.215
8x8C12	Zr	0.636	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.120	5.215

Table 6.2.1 (page 34 of 46)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640	60	0.4830	0.0320	0.4110	150	2 large/ 2 small	0.591/ 0.483	0.531/ 0.433	0.100	5.278
8x8D02	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.591	0.531	0.100	5.278
8x8D03	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.483	0.433	0.100	5.278
8x8D04	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.100	5.278
8x8D05	Zr	0.640	60	0.4830	0.0320	0.4100	150	1	1.34	1.26	0.100	5.278
8x8D06	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.120	5.278
8x8D07	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.080	5.278
8x8D08	*Zr*	*0.640*	*61*	*0.4830*	*0.0300*	*0.4140*	*150*	*3*	*0.591*	*0.531*	*0.080*	*5.278*
8x8E Assembly Class												
8x8E01	Zr	0.640	59	0.4930	0.0340	0.4160	150	5	0.493	0.425	0.100	5.278
8x8F Assembly Class												
8x8F01	*Zr*	*0.609*	*64*	*0.4576*	*0.0290*	*0.3913*	*150*	*4*†	*0.291*†	*0.228*†	*0.055*	*5.390*
9x9A Assembly Class												
9x9A01	Zr	0.566	74	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A02	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A03	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.100	5.278
9x9A04	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.120	5.278

† *Four rectangular water cross segments dividing the assembly into four quadrants*

BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9B Assembly Class												
9x9B01	Zr	0.569	72	0.4330	0.0262	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B02	Zr	0.569	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B03	*Zr*	*0.572*	*72*	*0.4330*	*0.0260*	*0.3737*	*150*	*1*	*1.516*	*1.459*	*0.100*	*5.278*
9x9C Assembly Class												
9x9C01	Zr	0.572	80	0.4230	0.0295	0.3565	150	1	0.512	0.472	0.100	5.278
9x9D Assembly Class												
9x9D01	Zr	0.572	79	0.4240	0.0300	0.3565	150	2	0.42~~5~~*4*	0.364	0.100	5.278
9x9E Assembly Class†												
9x9E01	Zr	0.572	76	0.4170	0.02~~90~~*65*	0.35~~25~~*30*	150	5	0.~~425~~*546*	0.~~364~~*522*	0.~~100~~*120*	5.2~~78~~*15*
9x9E02	*Zr*	*0.572*	*48* *28*	*0.4170* *0.4430*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*5*	*0.546*	*0.522*	*0.120*	*5.215*
9x9F Assembly Class†												
9x9F01	Zr	0.572	76	0.4430	0.0~~310~~*285*	0.3745	150	5	0.~~425~~*546*	0.~~364~~*522*	0.~~100~~*120*	5.2~~78~~*15*
9x9F02	*Zr*	*0.572*	*48* *28*	*0.4170* *0.4430*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*5*	*0.546*	*0.522*	*0.120*	*5.215*

† *The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in the CoC.*

Table 6.2.1 (page 6 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10A Assembly Class												
10x10A01	Zr	0.510	92	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A02	Zr	0.510	78	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A03	Zr	0.510	92/78	0.4040	0.0260	0.3450	155/90	2	0.980	0.920	0.100	5.278
10x10B Assembly Class												
10x10B01	Zr	0.510	91	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B02	Zr	0.510	83	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B03	Zr	0.510	91/83	0.3957	0.0239	0.3413	155/90	1	1.378	1.321	0.100	5.278
10x10C Assembly Class												
10x10C01	Zr	0.488	96	0.379~~90~~*80*	0.02*4*~~38~~	0.3224	150	5	~~0.4930~~ *1.227*	~~0.4250~~ *1.165*	0.055	5.457
10x10D Assembly Class												
10x10D01	SS	0.565	100	0.3960	0.0200	0.3500	83	0	n/a	n/a	0.08	5.663
10x10E Assembly Class												
10x10E01	SS	0.557	96	0.3940	0.0220	0.3430	83	4	0.3940	0.3500	0.08	5.663

Table 6.2.2 (page 1 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.527	0.493	0.0170
14x14A02	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.528	0.490	0.0190
14x14A03	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.526	0.492	0.0170
14x14B Assembly Class											
14x14B01	Zr	0.556	179	0.422	0.0243	0.3659	150	17	0.539	0.505	0.0170
14x14B02	Zr	0.556	179	0.417	0.0295	0.3505	150	17	0.541	0.507	0.0170
14x14B03	Zr	0.556	179	0.424	0.0300	0.3565	150	17	0.541	0.507	0.0170
14x14B04	Zr	0.556	179	0.426	0.0310	0.3565	150	17	0.541	0.507	0.0170
14x14C Assembly Class											
14x14C01	Zr	0.580	176	0.440	0.0280	0.3765	150	5	1.115	1.035	0.0400
14x14C02	Zr	0.580	176	0.440	0.0280	0.3770	150	5	1.115	1.035	0.0400
14x14C03	*Zr*	*0.580*	*176*	*0.440*	*0.0260*	*0.3805*	*150*	*5*	*1.111*	*1.035*	*0.0380*
14x14D Assembly Class											
14x14D01	SS	0.556	180	0.422	0.0165	0.3835	144	16	0.543	0.514	0.0145
15x15A Assembly Class											
15x15A01	Zr	0.550	204	0.418	0.0260	0.3580	150	5	0.533	0.500	0.0165

Table 6.2.2 (page 2 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.533	0.499	0.0170
15x15B02	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.546	0.512	0.0170
15x15B03	Zr	0.563	204	0.422	0.0243	0.3660	150	21	0.533	0.499	0.0170
15x15B04	Zr	0.563	204	0.422	0.0243	0.3659	150	21	0.545	0.515	0.0150
15x15B05	Zr	0.563	204	0.422	0.0242	0.3659	150	21	0.545	0.515	0.0150
15x15B06	Zr	0.563	204	0.420	0.0240	0.3671	150	21	0.544	0.514	0.0150
15x15C Assembly Class											
15x15C01	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.493	0.0255
15x15C02	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.511	0.0165
15x15C03	Zr	0.563	204	0.424	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15C04	Zr	0.563	204	0.417	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15D Assembly Class											
15x15D01	Zr	0.568	208	0.430	0.0265	0.3690	150	17	0.530	0.498	0.0160
15x15D02	Zr	0.568	208	0.430	0.0265	0.3686	150	17	0.530	0.498	0.0160
15x15D03	Zr	0.568	208	0.430	0.0265	0.3700	150	17	0.530	0.499	0.0155
15x15D04	Zr	0.568	208	0.430	0.0250	0.3735	150	17	0.530	0.500	0.0150
15x15E Assembly Class											
15x15E01	Zr	0.568	208	0.428	0.0245	0.3707	150	17	0.528	0.500	0.0140
15x15F Assembly Class											
15x15F01	Zr	0.568	208	0.428	0.0230	0.3742	150	17	0.528	0.500	0.0140

Table 6.2.2 (page 3 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563	204	0.422	0.0165	0.3825	144	21	0.543	0.514	0.0145
15x15H Assembly Class											
15x15H01	*Zr*	*0.568*	*208*	*0.414*	*0.0220*	*0.3622*	*150*	*17*	*0.528*	*0.500*	*0.0140*
16x16A Assembly Class											
16x16A01	Zr	0.506	236	0.382	0.0250	0.3255	150	5	0.980	0.900	0.0400
16x16A02	Zr	0.506	236	0.382	0.0250	0.3250	150	5	0.980	0.900	0.0400
17x17A Assembly Class											
17x17A01	Zr	0.496	264	0.360	0.0225	0.3088	144	25	0.474	0.442	0.0160
17x17A02	Zr	0.496	264	0.360	0.0225	0.3088	150	25	0.474	0.442	0.0160
17x17A03	Zr	0.496	264	0.360	0.0250	0.3030	150	25	0.480	0.448	0.0160
17x17B Assembly Class											
17x17B01	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.482	0.450	0.0160
17x17B02	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.474	0.442	0.0160
17x17B03	Zr	0.496	264	0.376	0.0240	0.3215	150	25	0.480	0.448	0.0160
17x17B04	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.427	0.399	0.0140
17x17B05	Zr	0.496	264	0.374	0.0240	0.3195	150	25	0.482	0.450	0.0160
17x17B06	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.480	0.452	0.0140
17x17C Assembly Class											
17x17C01	Zr	0.502	264	0.379	0.0240	0.3232	150	25	0.472	0.432	0.0200
17x17C02	Zr	0.502	264	0.377	0.0220	0.3252	150	25	0.472	0.432	0.0200

Table 6.2.6

MAXIMUM K_{EFF} VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24

(all dimensions are in inches)

14x14C (4.6% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2) 176 fuel rods, 5 guide tubes, pitch=0.580, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9361	0.9317	0.0009	0.440	0.3840	0.0280	0.3765	150	0.040
14x14C02	0.9355	0.9312	0.0008	0.440	0.3840	0.0280	0.3770	150	0.040
14x14C03	*0.9400*	*0.9357*	*0.0008*	*0.440*	*0.3880*	*0.0260*	*0.3805*	*150*	*0.038*
Dimensions Listed for Authorized Contents				0.440 (min.)	0.38~~40~~*80* (max.)		0.3~~770~~*805*† (max.)	150 (max.)	0.0~~40~~*38* (min.)
bounding dimensions (14x14C01)	0.9~~361~~*400*	0.9~~317~~*357*	0.000~~9~~*8*	0.440	0.38~~40~~*80*	0.02~~80~~*60*	0.3~~765~~*805*	150	0.0~~40~~*38*

~~† Because the k_{eff} values are statistically equivalent (within 1σ) for the small variation in pellet diameter, the pellet diameter listed in the specification of authorized contents is the larger of the two values.~~

Table 6.2.15
MAXIMUM K_{EFF} VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15H (3.8% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9411	0.9368	0.0008	0.414	0.3700	0.0220	0.3622	150	0.0140
Dimensions Listed for Authorized Contents				0.414 (min.)	0.3700 (max.)		0.3622 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.1~~5~~6
MAXIMUM K_{EFF} VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

16x16A (4.6% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9383	0.9339	0.0009	0.382	0.3320	0.0250	0.3255	150	0.0400
16x16A02	0.9371	0.9328	0.0008	0.382	0.3320	0.0250	0.3250	150	0.0400
Dimensions Listed for Authorized Contents				0.382 (min.)	0.3320 (max.)		0.3255 (max.)	150 (max.)	0.0400 (min.)
bounding dimensions (16x16A01)	0.9383	0.9339	0.0009	0.382	0.3320	0.0250	0.3255	150	0.0400

Table 6.2.167
MAXIMUM K_{EFF} VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17A (4.0% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17A01	0.9449	0.9400	0.0011	0.360	0.3150	0.0225	0.3088	144	0.016
17x17A02	0.9452†	0.9408	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016
17x17A03	0.9406	0.9364	0.0008	0.360	0.3100	0.0250	0.3030	150	0.016
Dimensions Listed for Authorized Contents				0.360 (min.)	0.3150 (max.)		0.3088 (max.)	150 (max.)	0.016 (min.)
bounding dimensions (17x17A02)	0.9452	0.9408	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016

† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9434.

Table 6.2.1~~7~~8
MAXIMUM K_{EFF} VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17B (4.0% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2)									
264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9377	0.9335	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B02	0.9379	0.9337	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B03	0.9330	0.9288	0.0008	0.376	0.3280	0.0240	0.3215	150	0.016
17x17B04	0.9407	0.9365	0.0007	0.372	0.3310	0.0205	0.3232	150	0.014
17x17B05	0.9349	0.9305	0.0009	0.374	0.3260	0.0240	0.3195	150	0.016
17x17B06	0.9436	0.9393	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014
Dimensions Listed for Authorized Contents				0.372 (min.)	0.3310 (max.)		0.3232 (max.)	150 (max.)	0.014 (min.)
bounding dimensions (17x17B06)	0.9436	0.9393	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014

Table 6.2.189
MAXIMUM K_{EFF} VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17C (4.0% Enrichment, Boral ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.502, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9383	0.9339	0.0008	0.379	0.3310	0.0240	0.3232	150	0.020
17x17C02	0.9427	0.9384	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020
Dimensions Listed for Authorized Contents				0.377 (min.)	0.3330 (max.)		0.3252 (max.)	150 (max.)	0.020 (min.)
bounding dimensions (17x17C02)	0.9427	0.9384	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020

Table 6.2.~~19~~*20*
MAXIMUM K_{EFF} VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

7x7B (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.738, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630	0.4990	0.0320	0.4870	150	n/a	0.080
7x7B02	0.9301	0.9260	0.0007	0.5630	0.4890	0.0370	0.4770	150	n/a	0.102
7x7B03	0.9313	0.9271	0.0008	0.5630	0.4890	0.0370	0.4770	150	n/a	0.080
7x7B04	0.9311	0.9270	0.0007	0.5700	0.4990	0.0355	0.4880	150	n/a	0.080
7x7B05	0.9350	0.9306	0.0008	0.5630	0.4950	0.0340	0.4775	150	n/a	0.080
7x7B06	*0.9298*	*0.9260*	*0.0006*	*0.5700*	*0.4990*	*0.0355*	*0.4910*	*150*	*n/a*	*0.080*
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.48~~80~~*910* (max.)	150 (max.)	n/a	0.120 (max.)
bounding dimensions (B7x7B01)	0.937~~8~~*5*	0.933~~5~~*2*	0.0008	0.5630	0.4990	0.0320	0.48~~80~~*910*	150	n/a	0.102
bounding dimensions with 120 mil channel (B7x7B02)	0.93~~75~~*86*	0.93~~32~~*44*	0.000~~8~~*7*	0.5630	0.4990	0.0320	0.48~~80~~*910*	150	n/a	0.120

Table 6.2.2~~0~~*1*

MAXIMUM K_{EFF} VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

8x8B (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 63 *or 64* fuel rods, 1 *or 0* water rod*s*, pitch† = 0.636-0.64~~1~~2, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	*Fuel rods*	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	*63*	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B02	0.9227	0.9185	0.0007	*63*	0.636	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B03	0.9299	0.9257	0.0008	*63*	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8B04	*0.9236*	*0.9194*	*0.0008*	*64*	*0.642*	*0.5015*	*0.4295*	*0.0360*	*0.4195*	*150*	*n/a*	*0.100*
Dimensions Listed for Authorized Contents				*63 or 64*	0.636-0.64~~1~~2	0.4840 (min.)	0.42~~50~~95 (max.)		0.41~~60~~95 (max.)	150 (max.)	0.034	0.120 (max.)
bounding (pitch=0.636) (B8x8B01)	0.93~~17~~*46*	0.9~~27~~*4301*	0.000~~8~~9	*63*	0.636	0.4840	0.42~~50~~95	0.02~~95~~725	0.41~~60~~95	150	0.034	0.120
bounding (pitch=0.640) (B8x8B02)	0.93~~57~~*85*	0.93~~15~~*43*	0.0008	*63*	0.640	0.4840	0.42~~50~~95	0.02~~95~~725	0.41~~60~~95	150	0.034	0.120
bounding (pitch=0.641) (B8x8B03)	0.93~~68~~*416*	0.93~~27~~*75*	0.0007	*63*	0.64~~1~~2	0.4840	0.42~~50~~95	0.02~~95~~725	0.41~~60~~95	150	0.034	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch *and a variation in the number of fuel and water rods.*

Table 6.2.2~~1~~2
MAXIMUM K_{EFF} VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

8x8C (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 62 fuel rods, 2 water rods, pitch† = 0.636-0.641, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8C02	0.9313	0.9268	0.0009	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.000
8x8C03	0.9329	0.9286	0.0008	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.800
8x8C04	0.9348††	0.9307	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.100
8x8C05	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.120
8x8C06	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8C07	0.9314	0.9273	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.100
8x8C08	0.9339	0.9298	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.034	0.100
8x8C09	0.9301	0.9260	0.0007	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8C10	0.9317	0.9275	0.0008	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C11	0.9328	0.9287	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C12	0.9285	0.9242	0.0008	0.636	0.4830	0.4190	0.0320	0.4110	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.636-0.641	0.4830 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9343.

Table 6.2.2~~2~~*3*

MAXIMUM K_{EFF} VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

8x8D (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 60 fuel rods, 1-4 water rods†, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830	0.4190	0.0320	0.4110	150	0.03/0.025	0.100
8x8D02	0.9325	0.9284	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8D03	0.9351	0.9309	0.0008	0.4830	0.4190	0.0320	0.4110	150	0.025	0.100
8x8D04	0.9338	0.9296	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.100
8x8D05	0.9339	0.9294	0.0009	0.4830	0.4190	0.0320	0.4100	150	0.040	0.100
8x8D06	0.9365	0.9324	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.120
8x8D07	0.9341	0.9297	0.0009	0.4830	0.4190	0.0320	0.4110	150	0.040	0.080
8x8D08	*0.9376*	*0.9332*	*0.0009*	*0.4830*	*0.4230*	*0.0300*	*0.4140*	*150*	*0.030*	*0.080*
Dimensions Listed for Authorized Contents				0.4830 (min.)	0.4~~190~~*230* (max.)		0.41~~10~~*40* (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B8x8D01)	0.9~~366~~*403*	0.93~~23~~*63*	0.0007~~8~~	0.4830	0.4~~190~~*230*	0.03~~20~~*00*	0.41~~10~~*40*	150	0.000	0.120

† Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. *Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.*

Table 6.2.234
MAXIMUM K_{EFF} VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

8x8E (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 59 fuel rods, 5 water rods, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
Dimensions Listed for Authorized Contents				0.4930 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.034 (min.)	0.100 (max.)

Table 6.2.25
MAXIMUM K_{EFF} VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

8x8F (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9140	0.9097	0.0008	0.4576	0.3996	0.0290	0.3913	150	0.0315	0.055
Dimensions Listed for Authorized Contents				0.4576 (min.)	0.3996 (max.)		0.3913 (max.)	150 (max.)	0.0315 (min.)	0.055 (max.)

Table 6.2.246
MAXIMUM K_{EFF} VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

9x9A (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2)
74/66 fuel rods†, 2 water rods, pitch=0.566, Zr clad

Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400	0.3840	0.0280	0.3760	150/90	0.030	0.100
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400	0.3840	0.0280	0.3760	150	0.030	0.120
Dimensions Listed for Authorized Contents				0.4400 (min.)	0.3840 (max.)		0.3760 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.000	0.120

† This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

Table 6.2.2~~5~~7
MAXIMUM K_{EFF} VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

9x9B (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 *to 0.572*†, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	*pitch*	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9368	0.9326	0.0007	*0.569*	0.4330	0.3807	0.0262	0.3737	150	0.0285	0.100
9x9B02	0.9377	0.9334	0.0008	*0.569*	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
9x9B03	*0.9416*	*0.9373*	*0.0008*	*0.572*	*0.4330*	*0.3810*	*0.0260*	*0.3737*	*150*	*0.0285*	*0.100*
Dimensions Listed for Authorized Contents				*0.572*	0.4330 (min.)	0.3810 (max.)		0.3740 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B9x9B01)	0.9~~388~~*422*	0.93~~46~~*80*	0.0007	*0.572*	0.4330	0.3810	0.0260	0.3740††	150	0.000	0.120

† *This assembly class was analyzed and qualified for a small variation in the pitch.*

†† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.268
MAXIMUM K_{EFF} VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

9x9C (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 80 fuel rods, 1 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230	0.3640	0.0295	0.3565	150	0.020	0.100
Dimensions Listed for Authorized Contents				0.4230 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.020 (min.)	0.100 (max.)

Table 6.2.279

MAXIMUM K_{EFF} VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

9x9D (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 79 fuel rods, 2 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.93924	0.934950	0.00089	0.4240	0.3640	0.0300	0.3565	150	0.03050	0.100
Dimensions Listed for Authorized Contents				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.03050 (min.)	0.100 (max.)

Table 6.2.~~28~~*30*

MAXIMUM K_{EFF} VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

9x9E (4.~~2~~*1*% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.94~~06~~*02*	0.93~~62~~*59*	0.0008	0.4170	0.3~~590~~*640*	0.02~~90~~*65*	0.35~~25~~*30*	150	0.0~~305~~*120*	0.~~100~~*120*
9x9E02	*0.9424*	*0.9380*	*0.0008*	*0.4170* *0.4430*	*0.3640* *0.3860*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*0.0120*	*0.120*
Dimensions Listed for Authorized Contents†				0.4170 (min.)	0.3~~590~~*640* (max.)		0.35~~25~~*30* (max.)	150 (max.)	0.0~~305~~*120* (min.)	0.~~100~~*120* (max.)
bounding dimensions (9x9E02)	*0.9424*	*0.9380*	*0.0008*	*0.4170* *0.4430*	*0.3640* *0.3860*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*0.0120*	*0.120*

† *This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for Authorized Contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Content lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.*

Table 6.2.~~29~~*31*

MAXIMUM K_{EFF} VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

9x9F (4.~~2~~*1*% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.93~~77~~*69*	0.93~~35~~*26*	0.0007	0.4430	0.38~~10~~*60*	0.0~~310~~*285*	0.3745	150	0.0~~305~~*120*	0.~~100~~*120*
9x9F02	*0.9424*	*0.9380*	*0.0008*	*0.4170* *0.4430*	*0.3640* *0.3860*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*0.0120*	*0.120*
Dimensions Listed for Authorized Contents†				0.4430 (min.)	0.38~~10~~*60* (max.)		0.3745 (max.)	150 (max.)	0.0~~305~~*120* (min.)	0.~~100~~*120* (max.)
bounding dimensions (9x9F02)	*0.9424*	*0.9380*	*0.0008*	*0.4170* *0.4430*	*0.3640* *0.3860*	*0.0265* *0.0285*	*0.3530* *0.3745*	*150*	*0.0120*	*0.120*

† *This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed for Authorized Contents, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Authorized Content lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.*

Table 6.2.302
MAXIMUM K_{EFF} VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

10x10A (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 92/78 fuel rods†, 2 water rods, pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040	0.3520	0.0260	0.3450	155/90	0.030	0.100
Dimensions Listed for Authorized Contents				0.4040 (min.)	0.3520 (max.)		0.3455 (max.)	150†† (max.)	0.030 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457†††	0.9414	0.0008	0.4040	0.3520	0.0260	0.3455‡	155	0.030	0.120

† This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for authorized contents limits the active fuel length to 150 inches. This is due to the fact that the Boral panels are 156 inches in length.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

‡ This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.3~~1~~3

MAXIMUM K_{EFF} VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10B (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 91/83 fuel rods†, 1 water rods (square, replacing 9 fuel rods), pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957	0.3480	0.0239	0.3413	155/90	0.0285	0.100
Dimensions Listed for Authorized Contents				0.3957 (min.)	0.3480 (max.)		0.3420 (max.)	150†† (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957	0.3480	0.0239	0.3420†††	155	0.000	0.120

† This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the specification for authorized contents limits the active fuel length to 150 inches. This is due to the fact that the Boral panels are 156 inches in length.

††† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.3~~2~~*4*

MAXIMUM K_{EFF} VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10C (4.2% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.~~8990~~*9021*	0.89~~49~~*80*	0.0007	0.37~~90~~*80*	0.3294	0.024~~3~~*8*	0.3224	150	0.03~~4~~*1*	0.055
Dimensions Listed for Authorized Contents				0.37~~90~~*80* (min.)	0.3294 (max.)		0.3224 (max.)	150 (max.)	0.03~~4~~*1* (min.)	0.055 (max.)

Table 6.2.335

MAXIMUM K_{EFF} VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68

(all dimensions are in inches)

10x10D (4.0% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 100 fuel rods, 0 water rods, pitch=0.565, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960	0.3560	0.0200	0.350	83	n/a	0.080
Dimensions Listed for Authorized Contents				0.3960 (min.)	0.3560 (max.)		0.350 (max.)	83 (max.)	n/a	0.080 (max.)

Table 6.2.346
MAXIMUM K_{EFF} VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches)

10x10E (4.0% Enrichment, Boral ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 4 water rods, pitch=0.557, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940	0.3500	0.0220	0.3430	83	0.022	0.080
Dimensions Listed for Authorized Contents				0.3940 (min.)	0.3500 (max.)		0.3430 (max.)	83 (max.)	0.022 (min.)	0.080 (max.)

Table 6.2.3~~5~~7

MAXIMUM K_{EFF} VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F

(all dimensions are in inches)

6x6A (3.0% Enrichment†, Boral ^{10}B minimum loading of 0.0067 g/cm^2) *35 or* 36 fuel rods††, *1 or* 0 water rods††, pitch=0.694 *to 0.710*††, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	*pitch*	*fuel rods*	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	*0.694*	*36*	0.5645	0.4945	0.0350	0.4940	110	n/a	0.060
6x6A02	0.7517	0.7476	0.0007	*0.694*	*36*	0.5645	0.4925	0.0360	0.4820	110	n/a	0.060
6x6A03	0.7545	0.7501	0.0008	*0.694*	*36*	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6A04	0.7537	0.7494	0.0008	*0.694*	*36*	0.5550	0.4850	0.0350	0.4820	110	n/a	0.060
6x6A05	*0.7555*	*0.7512*	*0.0008*	*0.696*	*36*	*0.5625*	*0.4925*	*0.0350*	*0.4820*	*110*	*n/a*	*0.060*
6x6A06	*0.7618*	*0.7576*	*0.0008*	*0.696*	*35*	*0.5625*	*0.4925*	*0.0350*	*0.4820*	*110*	*0.0*	*0.060*
6x6A07	*0.7588*	*0.7550*	*0.0007*	*0.700*	*36*	*0.5555*	*0.4850*	*0.03525*	*0.4780*	*110*	*n/a*	*0.060*
6x6A08	*0.7808*	*0.7766*	*0.0007*	*0.710*	*36*	*0.5625*	*0.5105*	*0.0260*	*0.4980*	*110*	*n/a*	*0.060*
Dimensions Listed for Authorized Contents				*0.710 (max.)*	*35 or 36*	0.5550 (min.)	~~0.4945~~ *0.5105* (max.)	*0.02225*	~~0.4940~~ *0.4980* (max.)	~~110~~ *120* (max.)	~~n/a~~*0.0*	0.060 (max.)
bounding dimensions (B6x6A01)	~~0.7602~~ *0.7727*	~~0.7562~~ *0.7685*	0.00067	*0.694*	*35*	0.5550	~~0.4945~~ *0.5105*	~~0.0303~~ *0.02225*	~~0.4940~~ *0.4980*	~~110~~ *120*	~~n/a~~*0.0*	0.060
bounding dimensions (B6x6A02)	*0.7782*	*0.7738*	*0.0008*	*0.700*	*35*	*0.5550*	*0.5105*	*0.02225*	*0.4980*	*120*	*0.0*	*0.060*
bounding dimensions (B6x6A03)	*0.7888*	*0.7846*	*0.0007*	*0.710*	*35*	*0.5550*	*0.5105*	*0.02225*	*0.4980*	*120*	*0.0*	*0.060*

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for authorized contents to 2.7%.

†† *This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.*

Table 6.2.~~36~~8
MAXIMUM K_{EFF} VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches)

6x6B (3.0% Enrichment†, Boral ^{10}B minimum loading of 0.0067 g/cm²) *35 or* 36 fuel rods†† *(up to* 9 MOX rods), *1 or* 0 water rods††, pitch=0.694 *to 0.710*††, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	*pitch*	*fuel rods*	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	~~0.7598~~ *0.7604*	~~0.7555~~ *0.7563*	0.00087	*0.694*	*36*	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6B02	~~0.7609~~ *0.7618*	~~0.7567~~ *0.7577*	0.0007	*0.694*	*36*	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B03	*0.7619*	*0.7578*	*0.0007*	*0.696*	*36*	*0.5625*	*0.4925*	*0.0350*	*0.4820*	*110*	*n/a*	*0.060*
6x6B04	*0.7686*	*0.7644*	*0.0008*	*0.696*	*35*	*0.5625*	*0.4925*	*0.0350*	*0.4820*	*110*	*0.0*	*0.060*
6x6B05	*0.7824*	*0.7785*	*0.0006*	*0.710*	*35*	*0.5625*	*0.4925*	*0.0350*	*0.4820*	*110*	*0.0*	*0.060*
Dimensions Listed for Authorized Contents				*0.710 (max.)*	*35 or 36*	0.5625 (min.)	0.4945 (max.)		0.4820 (max.)	~~110~~ *120* (max.)	~~n/a~~*0.0*	0.060 (max.)
bounding dimensions (B6x6B01)	~~0.7611~~ *0.7822*†††	~~0.7570~~ *0.7783*	0.0007	*0.710*	*35*	0.5625	0.4945	0.0340	0.4820	~~110~~ *120*	~~n/a~~*0.0*	0.060

Note:

1. These assemblies ~~consist of~~ *contain up to* 9 MOX pins~~and 27 UO_2 pins~~. The composition of the MOX fuel pins is given in Table 6.3.4.

† The ^{235}U enrichment of the MOX and UO_2 pins is assumed to be 0.711% and 3.0%, respectively.

†† *This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.*

††† *The k_{eff} value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1σ). Therefore, the 0.7824 value is listed in Tables 6.1.2 and 6.1.3 as the maximum.*

Table 6.2.3~~7~~9
MAXIMUM K_{EFF} VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches)

6x6C (3.0% Enrichment†, Boral ^{10}B minimum loading of 0.0067 g/cm^2) 36 fuel rods, 0 water rods, pitch=0.740, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630	0.4990	0.0320	0.4880	77.5	n/a	0.060
Dimensions Listed for Authorized Contents				0.5630 (min.)	0.4990 (max.)		0.4880 (max.)	77.5 (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for authorized contents to 2.7%.

Table 6.2.~~38~~*40*

MAXIMUM K_{EFF} VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F

(all dimensions are in inches)

7x7A (3.0% Enrichment†, Boral ^{10}B minimum loading of 0.0067 g/cm²) 49 fuel rods, 0 water rods, pitch=0.631, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.797~~3~~*4*	0.793~~0~~*2*	0.0008	0.4860	0.420~~0~~*4*	0.03~~30~~*28*	0.4110	~~7~~*980*	n/a	0.060
Dimensions Listed for Authorized Contents				0.4860 (min.)	0.420~~0~~*4* (max.)		0.4110 (max.)	~~7~~*980* (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for authorized contents to 2.7%.

Table 6.2.~~39~~*41*
MAXIMUM K_{EFF} VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches)

8x8A (3.0% Enrichment[†], Boral ^{10}B minimum loading of 0.0067 g/cm^2) *63 or* 64 fuel rods[††], 0 water rods, pitch=0.523, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	*fuel rods*	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	*64*	0.4120	0.3620	0.0250	0.3580	110	n/a	0.100
8x8A02	*0.7697*	*0.7656*	*0.0007*	*63*	*0.4120*	*0.3620*	*0.0250*	*0.3580*	*120*	*n/a*	*0.100*
Dimensions Listed for Authorized Contents				*63*	0.4120 (min.)	0.3620 (max.)		0.3580 (max.)	110 (max.)	n/a	0.100 (max.)
bounding dimensions (8x8A02)	*0.7697*	*0.7656*	*0.0007*	*63*	*0.4120*	*0.3620*	*0.0250*	*0.3580*	*120*	*n/a*	*0.100*

† Although the calculations were performed for 3.0%, the enrichment is limited in the specification for authorized contents to 2.7%.

†† *This assembly class was analyzed and qualified for a variation in the number of fuel rods.*

Table 6.2.42

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	*4.81"*
Canister Wall Thickness	*0.11"*
Separator Assembly Plates Thickness	*0.11"*
Cladding OD	*0.412"*
Cladding ID	*0.362"*
Pellet OD	*0.358"*
Active Length	*110.5"*
Fuel Composition	*1.8% UO_2 and 98.2% ThO_2*
Initial Enrichment	*93.5 wt% ^{235}U for 1.8% of the fuel*
Maximum k_{eff}	*0.1813*
Calculated k_{eff}	*0.1779*
Standard Deviation	*0.0004*

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

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.~~659~~*719*E-04	6.~~150~~*380*E-03
92238	2.285E-02	8.586~~4~~E-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

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

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 and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level).

6.4.4 Damaged Fuel Container

Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. *Two different DFC types with slightly different cross sections are analyzed.* DFCs containing fuel debris must be stored in the MPC-68F. 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

†† 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

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

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 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.

Table 6.4.8

MAXIMUM k_{eff} VALUES† IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Maximum†† k_{eff}	
	DFC Dimensions: ID 4.93" THK. 0.12"	*DFC Dimensions:ID 4.81" THK. 0.11"*
<u>6x6 Fuel Assembly</u>		
6x6 Intact Fuel	0.7086	*0.7016*
w/32 Rods Standing	0.7183	*0.7117*
w/28 Rods Standing	0.7315	*0.7241*
w/24 Rods Standing	0.7086	*0.7010*
w/18 Rods Standing	0.6524	*0.6453*
Collapsed to 8x8 array	0.7845	*0.7857*
Dispersed Powder	0.7628	*0.7440*
<u>7x7 Fuel Assembly</u>		
7x7 Intact Fuel	0.7463	*0.7393*
w/41 Rods Standing	0.7529	*0.7481*
w/36 Rods Standing	0.7487	*0.7444*
w/25 Rods Standing	0.6718	*0.6644*

† These calculations were performed with a planar-average enrichment of 3.0% and a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

†† Maximum k_{eff} includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED

FIGURE 6.4.1; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 4 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.2; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.3; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 12 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.4; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 18 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.5; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.6; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 13 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.7; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 24 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.8; FAILED FUEL CALCULATION MODEL (PLANAR CROSS-SECTION) WITH DAMAGED FUEL COLLAPSED INTO 8X8 ARRAY IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 6.4.10; THORIA ROD CANISTER (PLANAR CROSS-SECTION) WITH 18 THORIA RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

Table 6.C.1
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A01	0.9378	0.9332	0.0010	0.2147
14x14A02	0.9374	0.9328	0.0009	0.2137
14x14A03	0.9383	0.9340	0.0008	0.2125
14x14B01	0.9268	0.9225	0.0008	0.2788
14x14B02	0.9243	0.9200	0.0008	0.2398
14x14B03	0.9196	0.9152	0.0009	0.2598
14x14B04	0.9163	0.9118	0.0009	0.2631
B14x14B01	0.9323	0.9280	0.0008	0.2730
14x14C01	0.9361	0.9317	0.0009	0.2821
14x14C02	0.9355	0.9312	0.0008	0.2842
14x14C03	*0.9400*	*0.9357*	*0.0008*	*0.2900*
14x14D01	0.8576	0.8536	0.0007	0.3414
15x15A01	0.9301	0.9259	0.0008	0.2660
15x15B01	0.9427	0.9384	0.0008	0.2704
15C15B02	0.9441	0.9396	0.0009	0.2711
15x15B03	0.9462	0.9420	0.0008	0.2708
15x15B04	0.9452	0.9407	0.0009	0.2692
15x15B05	0.9473	0.9431	0.0008	0.2693
15x15B06	0.9448	0.9404	0.0008	0.2732
B15x15B01	0.9471	0.9428	0.0008	0.2722
15x15C01	0.9332	0.9290	0.0007	0.2563

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
15x15C02	0.9373	0.9330	0.0008	0.2536
15x15C03	0.9377	0.9335	0.0007	0.2525
15x15C04	0.9378	0.9338	0.0007	0.2499
B15x15C01	0.9444	0.9401	0.0008	0.2456
15x15D01	0.9423	0.9380	0.0008	0.2916
15x15D02	0.9430	0.9386	0.0009	0.2900
15x15D03	0.9419	0.9375	0.0009	0.2966
15x15D04	0.9440	0.9398	0.0007	0.3052
15x15E01	0.9475	0.9433	0.0007	0.2916
15x15F01	0.9478	0.9436	0.0008	0.3006
15x15G01	0.8986	0.8943	0.0008	0.3459
15x15H01	*0.9411*	*0.9368*	*0.0008*	*0.2425*
16x16A01	0.9383	0.9339	0.0009	0.2786
16x16A02	0.9371	0.9328	0.0008	0.2768
17x17A01	0.9449	0.9400	0.0011	0.2198
17x17A02	0.9452	0.9408	0.0008	0.2205
17x17A03	0.9406	0.9364	0.0008	0.2082
17x17B01	0.9377	0.9335	0.0008	0.2697
17x17B02	0.9379	0.9337	0.0008	0.2710
17x17B03	0.9330	0.9288	0.0008	0.2714
17x17B04	0.9407	0.9365	0.0007	0.2666
17x17B05	0.9349	0.9305	0.0009	0.2629

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
17x17B06	0.9436	0.9393	0.0008	0.2657
17x17C01	0.9383	0.9339	0.0008	0.2683
17x17C02	0.9427	0.9384	0.0008	0.2703

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6A01	0.7539	0.7498	0.0007	0.2754
6x6A02	0.7517	0.7476	0.0007	0.2510
6x6A03	0.7545	0.7501	0.0008	0.2494
6x6A04	0.7537	0.7494	0.0008	0.2494
6x6A05	*0.7555*	*0.7512*	*0.0008*	*0.2470*
6x6A06	*0.7618*	*0.7576*	*0.0008*	*0.2298*
6x6A07	*0.7588*	*0.7550*	*0.0005*	*0.2360*
6x6A08	*0.7808*	*0.7766*	*0.0007*	*0.2527*
B6x6A01	0.~~7602~~*7888*	0.~~7562~~*7846*	0.000~~6~~*7*	0.~~2677~~*2310*
6x6B01	0.~~7598~~*7604*	0.~~7555~~*7563*	0.000~~8~~*7*	0.~~2463~~*2461*
6x6B02	0.~~7609~~*7618*	0.~~7567~~*7577*	0.000~~7~~*6*	0.~~2461~~*2450*
6x6B03	*0.7619*	*0.7578*	*0.0007*	*0.2439*
6x6B04	*0.7686*	*0.7644*	*0.0008*	*0.2286*
6x6B05	*0.7824*	*0.7785*	*0.0006*	*0.2184*
B6x6B01	0.~~7611~~*7822*	0.~~7570~~*7783*	0.000~~7~~*6*	0.~~2442~~*2190*

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6C01	0.8021	0.7980	0.0007	0.2139
7x7A01	0.7973	0.7930	0.0008	0.2015
7x7B01	0.9372	0.9330	0.0007	0.3658
7x7B02	0.9301	0.9260	0.0007	0.3524
7x7B03	0.9313	0.9271	0.0008	0.3438
7x7B04	0.9311	0.9270	0.0007	0.3816
7x7B05	0.9350	0.9306	0.0008	0.3382
7x7B06	*0.9298*	*0.9260*	*0.0006*	*0.3957*
B7x7B01	0.~~9378~~*9375*	0.~~9335~~*9332*	0.0008	0.~~3794~~*3887*
B7x7B02	0.~~9375~~*9386*	0.~~9332~~*9344*	0.000~~8~~*7*	0.~~3839~~*3983*
8x8A01	0.7685	0.7644	0.0007	0.2227
8x8A02	*0.7697*	*0.7656*	*0.0007*	*0.2158*
8x8B01	0.9310	0.9265	0.0009	0.2935
8x8B02	0.9227	0.9185	0.0007	0.2993
8x8B03	0.9299	0.9257	0.0008	0.3319
8x8B04	*0.9236*	*0.9194*	*0.0008*	*0.3700*
B8x8B01	0.~~9317~~*9346*	0.~~9274~~*9301*	0.000~~8~~*9*	0.~~3319~~*3389*
B8x8B02	0.~~9357~~*9385*	0.~~9315~~*9343*	0.0008	0.~~3245~~*3329*
B8x8B03	0.~~9368~~*9416*	0.~~9327~~*9375*	0.0007	0.~~3231~~*3293*
8x8C01	0.9315	0.9273	0.0007	0.2822
8x8C02	0.9313	0.9268	0.0009	0.2716
8x8C03	0.9329	0.9286	0.0008	0.2877

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
8x8C04	0.9348	0.9307	0.0007	0.2915
8x8C05	0.9353	0.9312	0.0007	0.2971
8x8C06	0.9353	0.9312	0.0007	0.2944
8x8C07	0.9314	0.9273	0.0007	0.2972
8x8C08	0.9339	0.9298	0.0007	0.2915
8x8C09	0.9301	0.9260	0.0007	0.3183
8x8C10	0.9317	0.9275	0.0008	0.3018
8x8C11	0.9328	0.9287	0.0007	0.3001
8x8C12	0.9285	0.9242	0.0008	0.3062
B8x8C01	0.9357	0.9313	0.0009	0.3141
B8x8C02	0.9425	0.9384	0.0007	0.3081
B8x8C03	0.9418	0.9375	0.0008	0.3056
8x8D01	0.9342	0.9302	0.0006	0.2733
8x8D02	0.9325	0.9284	0.0007	0.2750
8x8D03	0.9351	0.9309	0.0008	0.2731
8x8D04	0.9338	0.9296	0.0007	0.2727
8x8D05	0.9339	0.9294	0.0009	0.2700
8x8D06	0.9365	0.9324	0.0007	0.2777
8x8D07	0.9341	0.9297	0.0009	0.2694
8x8D08	*0.9376*	*0.9332*	*0.0009*	*0.2841*
B8x8D01	0.~~9366~~*9403*	0.~~9323~~*9363*	0.000~~8~~7	0.~~2740~~2778
8x8E01	0.9312	0.9270	0.0008	0.2831

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
8x8F01	*0.9140*	*0.9097*	*0.0008*	*0.2505*
9x9A01	0.9353	0.9310	0.0008	0.2875
9x9A02	0.9388	0.9345	0.0008	0.2228
9x9A03	0.9351	0.9310	0.0007	0.2837
9x9A04	0.9396	0.9355	0.0007	0.2262
B9x9A01	0.9417	0.9374	0.0008	0.2236
9x9B01	0.9368	0.9326	0.0007	0.2561
9x9B02	0.9377	0.9334	0.0008	0.2547
9x9B03	*0.9416*	*0.9373*	*0.0008*	*0.2517*
B9x9B01	0.~~9388~~*9422*	0.~~9346~~*9380*	0.0007	0.~~2530~~*2501*
9x9C01	0.9395	0.9352	0.0008	0.2698
9x9D01	0.~~9392~~*9394*	0.~~9349~~*9350*	0.000~~8~~*9*	0.~~2628~~*2625*
9x9E01	0.~~9406~~*9402*	0.~~9362~~*9359*	0.0008	0.~~2283~~*2249*
9x9E02	*0.9424*	*0.9380*	*0.0008*	*0.2088*
9x9F01	0.~~9377~~*9369*	0.~~9335~~*9326*	0.000~~7~~*8*	0.~~3028~~*2954*
9x9F02	*0.9424*	*0.9380*	*0.0008*	*0.2088*
10x10A01	0.9377	0.9335	0.0008	0.3170
10x10A02	0.9426	0.9386	0.0007	0.2159
10x10A03	0.9396	0.9356	0.0007	0.3169
B10x10A01	0.9457	0.9414	0.0008	0.2212
10x10B01	0.9384	0.9341	0.0008	0.2881
10x10B02	0.9416	0.9373	0.0008	0.2333

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
10x10B03	0.9375	0.9334	0.0007	0.2856
B10x10B01	0.9436	0.9395	0.0007	0.2366
10x10C01	0.~~8990~~*9021*	0.~~8949~~*8980*	0.0007	0.~~2656~~*2610*
10x10D01	0.9376	0.9333	0.0008	0.3355
10x10E01	0.9185	0.9144	0.0007	0.2936

Note: Maximum k_{eff} = Calculated k_{eff} + $K_c \times \sigma_c$ + Bias + σ_B
where:

K_c = 2.0
σ_c = Std. Dev. (1-sigma)
Bias = 0.0021
σ_B = 0.0006

See Subsection 6.4.3 for further explanation.

CHAPTER 7: OPERATING PROCEDURES

7.0 INTRODUCTION

This chapter outlines the procedures for loading, preparation for shipment, unloading, and preparation for empty cask shipment of the HI-STAR 100 System in accordance with 10CFR71 [7.0.1]. The procedures provided in this chapter are prescriptive in that they provide the basis and general guidance for plant personnel in preparing detailed written site-specific loading, handling, and unloading procedures. Users may add or delete steps in their site-specific implementation procedures provided the intent of these guidelines is met. Section 7.1 provides the guidance for loading the HI-STAR 100 System in the spent fuel pool. Section 7.2 provides the guidance for unloading the HI-STAR 100 System in the spent fuel pool. Section 7.3 provides the guidance for the preparation of the empty HI-STAR 100 for transport. Section 7.4 provides guidance for preparing the HI-STAR 100 Overpack for transport following a period of storage. Equipment specific operating details such as Vacuum Drying System valve manipulation and onsite transporter operation will be provided to users based on the specific equipment selected by the users and the configuration of the site.

Licensees (Users) will utilize the procedures provided in this chapter, the conditions of the Certificate of Compliance, equipment-specific operating instructions, and plant working procedures and apply them to develop the site-specific written loading, handling, unloading and storage procedures. The procedures contained herein describe acceptable methods for performing HI-STAR 100 loading and unloading operations. Users may alter these procedures to allow operations to be performed in parallel or out of sequence as long as the general intent of the procedure is met. In the figures following each section, acceptable configurations of rigging, piping, equipment and instrumentation are shown. Users may select alternate equipment, configurations and methodology to accommodate their specific needs. Any deviations to the rigging should be approved by the user's load handling authority.

The loading and unloading procedures in Section 7.1 and 7.2 can also be appropriately revised into written site-specific procedures to allow dry loading and unloading of the system in a hot cell or other remote handling facility. The Dry Transfer Facility (DTF) loading and unloading procedures are essentially the same with respect to loading and vacuum drying, inerting, and leakage testing of the MPC. Section 7.4 provides a synopsis of the regulatory requirements for the HI-STAR 100 package. The dry transfer facility shall develop the appropriate site-specific procedures as part of the DTF facility license.

Tables 7.1.1 and 7.1.2, respectively provide the handling weights for each of the HI-STAR 100 System major components and the loads to be lifted during the operation of the HI-STAR 100 System. Table 7.1.3 provides the HI-STAR 100 System bolt torque and sequencing requirements. Table 7.1.4 provides an operational description of the HI-STAR 100 System ancillary equipment and its safety designation. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in the Certificate of Compliance are loaded into the HI-STAR 100 System.

Users will be required to develop or modify existing programs and procedures to account for the transport operation of the HI-STAR 100 and future potential storage at an ISFSI. Written procedures will be required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of items and components identified as Important to Safety, heavy load handling, specialized instrument calibration, special nuclear material accountability, fuel handling procedures, training, equipment and process qualifications. Users shall implement controls to ensure that the lifted weights do not exceed the HI-STAR 100 trunnion design limits. Users shall implement controls to monitor the time limit from the removal of the HI-STAR 100 from the spent fuel pool to the commencement of MPC draining to prevent boiling. Chapter 3 of this SAR provides *examples of* ~~the~~ time limits based on *representative* ~~the~~ spent fuel pool temperatures *and design basis heat loads*. Users shall also implement controls to ensure that the HI-STAR 100 overpack cannot be subjected to a fire in excess of design limits during loading operations.

Table 7.1.5 summarizes the instrumentation necessary to load and unload the HI-STAR 100 System. Tables 7.1.6 and 7.1.7 provide sample receipt inspection checklists for the HI-STAR 100 overpack and the MPC, respectively. Users shall develop site-specific receipt inspection checklists, as required. Fuel handling, including the handling of fuel assemblies in the Damaged Fuel Container (DFC) shall be performed in accordance with written site-specific procedures. *Damaged fuel and fuel debris, as defined in the CoC, shall be loaded in* DFCs ~~shall be loaded in the spent fuel pool racks prior to placement into the MPC~~.

7.0.1 Technical and Safety Basis for Loading and Unloading Procedures

The procedures herein (7.1 through 7.3) are developed for the loading, unloading, and empty *(after initial transport)* transport of the HI-STAR 100 System. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present personnel hazards and radiological impact. The design of the HI-STAR 100 System, including these procedures and the ancillary equipment to minimize risks and mitigate consequences of potential events. The primary objective is to reduce the risk of occurrence and/or to mitigate the consequences of the event. The procedures contain Notes, Warnings, and Cautions to notify the operators to upcoming situations and provide additional information as needed. The Notes, Warnings and Cautions are purposely bolded and boxed, and immediately precede the applicable steps.

In the event of an extreme abnormal condition (e.g., cask drop or tip-over event) the user shall have appropriate procedural guidance to respond to the situation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusion zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution.

7.1 PROCEDURE FOR LOADING THE HI-STAR 100 SYSTEM IN THE SPENT FUEL POOL AND PREPARATION FOR SHIPMENT

7.1.1 Overview of Loading Operations

The HI-STAR 100 System is used to load and transport spent fuel. Specific steps are performed to prepare the HI-STAR 100 System for fuel loading, to load the fuel, to prepare the system for transport and to ship the HI-STAR 100 System. The HI-STAR 100 overpack may be transported off-site using a rail car or a specially designed heavy haul trailer, or any other load handling equipment designed for such applications. Users shall develop detailed written procedures to control on-site transport operations. Section 7.1.2 provides the general procedures for handling of the HI-STAR 100 overpack and MPC.

Figure 7.1.1 shows a flow diagram of the HI-STAR 100 System loading operations. Figure 7.1.2 illustrates some of the major HI-STAR 100 System loading operations. The HI-STAR 100 overpack and empty MPC may arrive together or separately. The procedures provided assume that these components arrive separately. If the HI-STAR 100 overpack and MPC arrive together, certain steps of the procedure may be omitted.

Note:
The procedures describe plant facilities, functions, and processes in general terms. Each site is different with regard to layout, organization and nomenclature. Users shall interpret the nomenclature used herein to suit their particular site, organization, and methods of operation.

Refer to the boxes of Figure 7.1.2 for the following description. The HI-STAR 100 overpack is received and the personnel barrier is removed. Receipt inspection and radiological surveys are performed. The impact limiters are removed and the HI-STAR 100 overpack is upended. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into the HI-STAR 100 overpack (Box 2). The annulus is filled with plant demineralized water and the MPC is filled with either spent fuel pool water or plant demineralized water (Box 3). An inflatable seal is installed in the annulus between the MPC and the HI-STAR 100 overpack to prevent spent fuel pool water from contaminating the exterior surface of the MPC. The HI-STAR 100 overpack and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the lid retention system (Box 6). The lift yoke remotely engages to the HI-STAR 100 overpack lifting trunnions to lift the HI-STAR 100 overpack and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-STAR 100 overpack from the spent fuel pool, the cask is removed from the spent fuel pool. If the lid retention system is being used, the HI-STAR 100 overpack closure plate bolts are installed to the lid retention disk to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-STAR 100 overpack are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

The HI-STAR 100 overpack is placed in the designated preparation area and the lift yoke and lid retention system retention disk are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of the HI-STAR 100 overpack are decontaminated. The Temporary Shield Ring (if utilized) is installed and filled with water. The inflatable annulus seal is removed, and the annulus shield is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-STAR 100 overpack during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid and around the mid-height circumference of the HI-STAR 100 overpack to establish appropriate radiological control.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the Automated Welding System (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant examinations are performed on the root and final passes. A volumetric examination is performed on the MPC welds to ensure that the completed weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The water level is raised to the top of the MPC and a hydrostatic test is performed on the MPC Lid-to-Shell welds to verify structural integrity. A small amount of water is displaced with helium gas for leakage testing. A leakage rate test is performed on the MPC lid-to-shell weld to verify weld integrity and to ensure that leakage rates are within acceptance criteria.

The ~~water level is raised to the top of the MPC again and then the~~ MPC water is displaced from the MPC by blowdown of the water using pressurized helium or nitrogen gas introduced into the vent port of the MPC thus displacing the water through the drain line. ~~The volume of water displaced from the MPC is measured to determine the free volume inside the MPC. This information is used later in the operation to determine the helium backfill requirements for the MPC.~~

The Vacuum Drying System is connected to the MPC and is used to remove all liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System. The internal pressure is reduced to below 3 torr and held for 30 minutes to ensure that all liquid water is removed.

Following the dryness test, the vacuum drying system is disconnected, the Helium Backfill System is connected, and the MPC is backfilled with a predetermined ~~mass~~ *pressure* of helium gas. The helium backfill ensures adequate heat transfer during transport, and provides the means of future leakage rate testing of the MPC confinement boundary welds for future storage. Cover plates are installed and seal welded over the MPC vent and drain ports and liquid penetrant examinations are performed on the root *(for multi-pass welds)* and final passes (Box 10). The cover plates are leakage tested to confirm that they meet the established leakage rate criteria.

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root *(for multi-pass welds)* and final welds 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 Temporary Shield Ring is drained and removed. The MPC lid and accessible areas at the top of the MPC shell are smeared for removable contamination. The HI-STAR 100 overpack closure plate is installed (Box 11) and the bolts are torqued. The HI-STAR 100 overpack annulus is dried and backfilled with helium gas. The HI-STAR 100 overpack mechanical seals are leakage tested to assure they will provide long-term retention of the annulus helium. The HI-STAR 100 overpack vent and drain port cover plates are installed. The HI-STAR 100 overpack is surveyed for removable contamination.

The HI-STAR 100 overpack is moved to the transport location. The HI-STAR 100 is downended on the transport frame. An inspection for signs of impaired condition is performed. Contamination surveys are performed. The HI-STAR 100 overpack is placed on the transport device, the tie-down and impact limiters are installed and a shielding effectiveness test is performed to ensure that the HI-STAR 100 shielding has been manufactured and is functioning as designed. Radiation levels are verified to be within acceptable limits. The assembled package is given a final inspection to verify that all conditions for transport have been met (e.g., all mechanical seals have been installed and tested, rupture disks are intact, installed and not covered. The carrier is provided with the appropriate paperwork and the receiver is notified of the impending shipment) and the personnel barrier is installed (Box 12). The package is then labeled, placarded and released for transport.

7.1.2 HI-STAR 100 System Receiving and Handling Operations:

Note:
The HI-STAR 100 overpack may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for the HI-STAR 100 overpack and MPC rigging and handling. Site-specific procedures shall specify the required operational sequences based on the cask handling configuration and limitations at the sites.

Note:
Steps 1 through 4 describe the handling operations using a lift yoke. Specialty rigging may be substituted if the lift complies with NUREG-0612 [7.1.1].

1. Vertical Handling of the HI-STAR 100 overpack:

Note:
Prior to performing any lifting operation, the removable shear ring segments under the two lifting trunnions must be removed.

Caution:
Users shall maintain controls to ensure that heights to which the loaded HI-STAR 100 is lifted outside the fuel building ~~is limited~~ are limited to ensure that the structural integrity of the MPC and overpack is not compromised should the overpack be dropped. This also assumes the *on-site* surfaces over which the loaded overpack will be transported have been designed and constructed consistent with the analysis assumptions provided in the HI-STAR 100 Topical Safety Analysis Report [7.1.2].

a. Verify that the lift yoke load test certifications are current.

b. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage.

c. Engage the lift yoke to the lifting trunnions. See Figure 7.1.3.

d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

Note:
Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements.

e. Raise the HI-STAR 100 overpack and position it accordingly.

2. Upending of the HI-STAR 100 overpack in the transport frame:

Warning:
Personnel shall remain clear of the unshielded bottom of the loaded overpack. Users shall coordinate operations to keep the bottom cover installed to the maximum extent practicable whenever when the loaded overpack is downended.

a. Position the HI-STAR 100 overpack under the lifting device. Refer to Step 1, above.

b. Verify that the lift yoke load test certifications are current.

c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage.

d. Place a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent) on the cask trunnions and the palms of the lift yoke.

e. Engage the lift yoke to the lifting trunnions. See Figure 7.1.3.

f. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

g. Slowly rotate the HI-STAR 100 overpack to the vertical position keeping all rigging as close to vertical as practicable. See Figure 7.1.4.

h. Lift the pocket trunnions clear of the transport frame rotation trunnions.

i. Position the HI-STAR 100 overpack per site direction.

3. Downending of the HI-STAR 100 overpack in the transport frame:

a. Position the transport frame under the lifting device.

b. Verify that the lift yoke load test certifications are current.

c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage.

d. Place a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent) on the cask trunnions and the palms of the lift yoke.

e. Place a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent) in the inside surfaces of the cask rotation trunnion pockets and the corresponding surfaces of the transport frame.

f. Engage the lift yoke to the lifting trunnions. See Figure 7.1.3.

g. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.

h. Position the pocket trunnions to receive the transport frame rotation trunnions. See Figure 7.1.4.

i. Slowly rotate the HI-STAR 100 overpack to the horizontal position keeping all rigging as close to vertical as practicable.

j. Disengage the lift yoke.

Warning:
Personnel shall remain clear of the unshielded bottom of the loaded overpack. Users shall coordinate operations to keep the bottom cover installed to the maximum extent practicable whenever when the loaded overpack is downended.

k. If necessary for radiation shielding, install the overpack bottom cover. Rigging points are provided. See Figure 7.1.4.

4. Horizontal Handling of the HI-STAR 100 overpack in the transport frame:

a. Secure the transport frame for HI-STAR 100 downending.

b. Downend the HI-STAR 100 overpack on the transport frame per Step 3, if necessary.

c. Inspect the transport frame lift rigging in accordance with site approved rigging procedures.

d. Position the transport frame accordingly.

5. Empty MPC Installation in the HI-STAR 100 overpack:

> **Note:**
> To avoid side loading the MPC lift lugs, the MPC must be upended in the MPC Upending Frame *(or equivalent)*. See Figure 7.1.6.

a. If necessary, remove any MPC shipping covers and rinse off any road dirt with water. Be sure to remove any foreign objects from the MPC internals.

b. Upend the MPC as follows:

1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage.

2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 7.1.6.

> **Warning:**
> The Upending Frame rigging bars are equipped with cleats that prevent the slings from sliding along the bar. The slings must be placed to the outside of the cleats to prevent an out-of-balance condition. The Upending Frame rigging points are labeled.

3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 7.1.6.

4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device).

5. Raise the MPC in the Upending Frame.

> **Warning:**
> The Upending Frame corner should be kept close to the ground during the upending process.

6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.

7. When the MPC approaches the vertical orientation, release the tension on the lower slings.

8. Place the MPC in a vertical orientation on a level surface.

9. Disconnect the MPC straps and disconnect the rigging.

c. Install the MPC in the HI-STAR 100 overpack as follows:

1. Install the four point lift sling to the lift lugs inside the MPC. See Figure 7.1.7.

Caution:
Be careful not to damage the seal seating surface during MPC installation.

2. Raise and place the MPC inside the HI-STAR 100 overpack.

Note:
An alignment punch mark is provided on the HI-STAR 100 overpack and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring. See Figure 7.1.8.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside the HI-STAR 100 overpack. Disconnect the MPC rigging or the MPC lift rig.

7.1.3 HI-STAR 100 Overpack and MPC Receipt Inspection and Loading Preparation

1. Recover the shipping documentation from the carrier.

   a. If necessary, recover the keys to the personnel barrier locks from the carrier.

   b. Record the impact limiter security seal serial numbers and verify that they match the corresponding shipping documentation, as applicable.

   c. Perform a receipt radiation and contamination survey in accordance with 49CFR173.443 [7.1.3] and 10CFR20.1906 [7.1.4].

2. If necessary, remove the personnel barrier as follows:

Note:
The personnel barrier is a ventilated enclosure cage that fits over the main body of the HI-STAR 100 overpack. The personnel barrier is designed to restrict personnel accessibility to the potentially hot surfaces of the HI-STAR 100 overpack. The personnel barrier in conjunction with the impact limiters restrict accessibility to all surfaces of the HI-STAR 100 overpack during transport. The personnel barrier is equipped with locks to prevent unauthorized access.

   a. Remove the locks securing the personnel barrier and remove the personnel barrier. Lifting points and a small bridle sling is provided. See Figure 7.1.9.

   b. Remove the pins securing the personnel barrier to the transport frame.

   c. Rig the personnel barrier to the lifting device as shown on Figure 7.1.9.

   d. Remove the personnel barrier.

e. Perform a partial visual inspection of the overpack surfaces to verify that there is no outward indication that would suggest impaired condition of the overpack in accordance with 10CFR71.87(b) [7.0.1]. Identify any significant indications to the cognizant individual for evaluation and resolution and record on the receiving documentation.

3. If necessary, remove the impact limiters as follows:

Note:
To prevent damage to the impact limiters the impact limiter handling frame must be used to remove, install, handle and store the impact limiters.

a. Clip the security seal wires and remove the security seals and wires.

b. Attach the impact limiter handling frame as shown on Figure 7.1.10. The rigging arms secure the impact limiter and maintain it at the proper orientation during rigging.

Caution:
The slings should be preloaded to the impact limiter weight plus the weight of the impact limiter handling frame prior to removal of the impact limiter bolts. (See Table 7.1.1) This will prevent damage to the HI-STAR 100 overpack and impact limiter from excessive lift pressure during removal.

c. Using a load measuring device, apply the correct lift load. See Table 7.1.1 for weights.

d. Remove the bolts securing the impact limiter to the overpack. See Figure 7.1.11.

e. Remove the impact limiter and store the impact limiter and bolts in a site-approved location.

f. Repeat Steps 3.c. through 3.e. for the other impact limiter.

g. Remove the alignment pins from the bottom of the HI-STAR 100 overpack. See Figure 7.1.11.

h. Complete the visual inspection to verify that there is no outward indication that would suggest impaired condition of the overpack. (10CFR71.87(b)) [7.0.1]. Identify any significant indications to the cognizant individual for evaluation and resolution.

i. Verify that the HI-STAR 100 overpack neutron shield rupture discs are installed, intact and not covered by tape or other covering.

ALARA Note:
A bottom protective cover may be attached to the HI-STAR 100 overpack bottom or placed in the designated preparation area or spent fuel pool. This will help prevent imbedding contaminated particles in the HI-STAR 100 overpack bottom surface and ease the decontamination effort.

4. Place the HI-STAR 100 overpack in the cask receiving area.

5. If necessary, remove the buttress plate bolts and remove the buttress plate. See Figure 7.1.11. See Figure 7.1.12 for rigging. Store these components in a site-approved storage location.

6. If necessary, remove the HI-STAR 100 overpack closure plate by removing the closure plate bolts and using the dedicated lift sling. See Figure 7.1.12 for rigging.

   a. Place the closure plate on cribbing that protects the seal seating surfaces and allows access for seal replacement.

   b. Store the closure plate and bolts in a site-approved location.

   c. Install the seal surface protector on the HI-STAR 100 overpack seal seating surface. See Figure 7.1.13.

7. Install the MPC inside the HI-STAR 100 overpack as follows:

   a. Rinse off any MPC road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.

   b. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

   c. Place the HI-STAR 100 overpack in the designated preparation area.

Note:
Upper fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. See Figure 7.1.14. *Upper Fuel spacers may be loaded any time prior to placement of the MPC lid in the spent fuel pool for installation in the MPC.*

8. Install the upper fuel spacers in the MPC lid as follows:

Warning:
Never work under a suspended load.

   a. Position the MPC lid on supports to allow access to the underside of the MPC lid.

   b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Figure 7.1.14 and Table 7.1.3 for torque requirements.

c. Install threaded plugs in the MPC lid where and when spacers will not be installed, if necessary. See Table 7.1.3 for torque requirements.

9. At the user's discretion, perform an MPC lid and closure ring fit test:

Note:
It will be necessary to perform the MPC installation and inspection in a location that has sufficient crane clearance to perform the operation.

a. Visually inspect the MPC lid rigging (See Figure 7.1.12).

b. Raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid. See Figure 7.1.15.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. See Figure 7.1.16. Install the MPC lid. Verify that the MPC lid fit and weld prep are in accordance with the approved design drawings.

ALARA Note:
The closure ring is installed by hand. No tools are required.

d. Install the closure ring.

e. Verify that closure ring fit and weld prep are in accordance with the approved design drawings.

f. Remove the closure ring and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.

Note:
Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used. *Fuel spacers may be loaded any time prior to insertion of the fuel assemblies in the MPC.*

10. Install lower fuel spacers in the MPC (if required for the fuel type). See Figure 7.1.14.

11. Fill the MPC and annulus as follows:

Caution:
Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

a. Remove the HI-STAR 100 overpack drain port cover and port plug and install the drain connector. Store the drain port cover plate and port plug in an approved storage location.

b. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

c. Manually insert the inflatable annulus seal around the MPC. See Figure 7.1.13.

d. Ensure that the seal is uniformly positioned in the annulus area.

e. Inflate the seal to between 30 and 35 psig *or as directed by the manufacturer.*

f. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

ALARA Note:
Waterproof tape placed over empty bolt holes, and bolt plugs may reduce the time required for decontamination.

12. At the user's discretion, install the HI-STAR 100 overpack closure plate bolt plugs and/or apply waterproof tape over any empty bolt holes.

ALARA Note:
Keeping the water level below the top of the MPC prevents splashing during handling.

13. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches below the top of the MPC shell.

14. Place the HI-STAR 100 overpack in the spent fuel pool as follows:

ALARA Note:
The Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations. The Annulus Overpressure System is equipped with double-locking quick disconnects to prevent inadvertent draining. The reservoir valve must be closed to ensure that the annulus is not inadvertently drained through the Annulus Overpressure System when the cask is raised above the level of the annulus reservoir.

a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-STAR 100 overpack via the quick disconnect. See Figure 7.1.17.

b. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions and position the HI-STAR 100 overpack over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

c. Wet the surfaces of the HI-STAR 100 overpack and lift yoke with plant demineralized water while slowly lowering the HI-STAR 100 overpack into the spent fuel pool.

d. When the top of the HI-STAR 100 overpack reaches the elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

e. Place the HI-STAR 100 overpack on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged. Remove the lift yoke from the spent fuel pool while spraying the crane cables and yoke with plant demineralized water.

7.1.4 MPC Fuel Loading

Note:
An underwater camera or other suitable viewing device may be used for monitoring underwater operations.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading as specified in the Certificate of Compliance have been selected for loading into the MPC.

2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.

3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

7.1.5 MPC Closure

Note:
The user may elect to use the optional Lid Retention System (See Figure 7.1.18) to assist in the installation of the MPC lid and attachment of the lift yoke, and to provide the means to secure the MPC lid in the event of a drop or tip-over accident during loaded cask handling operations outside of the spent fuel pool. The user is responsible for evaluating the additional weight imposed on the cask, lift yoke, crane and floor prior to use to ensure that its use does not exceed the crane capacity, heavy loads handling restrictions, or 250,000 pounds. See Tables 7.1.1 and 7.1.2.

1. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.

2. Install the drain line to the underside of the MPC lid. See Figure 7.1.15.

3. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage. See Figure 7.1.16 and 7.1.19.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

4. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented. See Figure 7.1.16.

5. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

Note:
The upper surface of the MPC lid will seat approximately flush with the top edge of the MPC shell when properly installed.

6. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

7. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions.

8. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

ALARA Note:
Activated debris may have settled on the top face of the HI-STAR 100 overpack and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal.

9. Raise the HI-STAR 100 overpack until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Raise and flush the upper surface of the HI-STAR 100 overpack and MPC with the plant demineralized water hoses as necessary to remove any activated particles from the HI-STAR 100 overpack or the MPC lid.

10. Visually verify that the MPC lid is properly seated. Lower the HI-STAR 100 overpack, reinstall the MPC lid, and repeat Step 9, as necessary.

11. If the Lid Retention System is used, inspect the closure plate bolts for general condition. Replace worn or damaged bolts with new bolts.

12. Install the Lid Retention System bolts if the Lid Retention System is used.

Warning:
Cask removal from the spent fuel pool is the heaviest lift that occurs during HI-STAR 100 loading operations. The HI-STAR 100 trunnions must not be subjected to lifted loads in excess of 250,000 lbs.. Users must ensure that plant-specific lifting equipment is qualified to lift the expected load. Users may elect to pump a measured quantity of water from the MPC prior to removing the HI-STAR 100 from the spent fuel pool. See Table 7.1.1 and 7.1.2 for weight information.

13. If necessary for lifted weight conditions, pump a measured amount of water from the MPC. See Figure 7.1.22 and Tables 7.1.1 and 7.1.2.

14. Continue to raise the HI-STAR 100 overpack under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-STAR 100 overpack reaches the approximate elevation as the reservoir, close the Annulus Overpressure System reservoir valve. See Figure 7.1.17.

Caution:
Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 3 of this SAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action *if time limits are approached or exceeded.* One course of action involves initiating an MPC water flush for a certain duration and flow rate. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded.

ALARA Note:
To reduce decontamination time, the surfaces of the HI-STAR 100 overpack and lift yoke should be kept wet until decontamination begins.

15. Remove the HI-STAR 100 overpack from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

ALARA Note:
Decontamination of the HI-STAR 100 overpack bottom should be performed using pole-mounted cleaning devices.

16. Decontaminate the HI-STAR 100 overpack bottom and perform a contamination survey of the HI-STAR 100 overpack bottom. Remove the bottom protective cover, if used.

17. If used, disconnect the Annulus Overpressure System from the HI-STAR 100 overpack via the quick disconnect. See Figure 7.1.17.

18. Set the HI-STAR 100 overpack in the designated cask preparation area.

19. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

Warning:
MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the 429 mrem/hour could indicate that fuel assemblies not meeting *the specifications in the CoC have been loaded*~~Technical Specifications~~.

a. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose rate is below 429 mrem/hour.

20. Perform decontamination of the HI-STAR 100 overpack.

21. Prepare the MPC for MPC lid welding as follows:

ALARA Note:
The Temporary Shield Ring is installed by hand, no tools are required.

a. Decontaminate the area around the HI-STAR 100 overpack top flange and install the Temporary Shield Ring, (if used). See Figure 7.1.20.

b. Fill the Temporary Shield Ring with water (if used).

c. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable annulus seal.

d. Deflate and remove the annulus seal.

ALARA Note:
The water in the HI-STAR 100 overpack-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

22. Attach the drain line to the HI-STAR 100 overpack drain port connector and lower the annulus water level approximately 6 inches.

ALARA Note:
The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded. Removable contamination on the exterior surfaces of the Overpack and accessible portions of the MPC shall each not exceed:

a. 2200 dpm/100 cm^2 from beta and gamma sources; and

b. 220 dpm/100 cm^2 from alpha sources.

a. Survey the MPC lid top surfaces and the accessible areas of the top two inches of the MPC shell.

ALARA Note:
The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

23. Install the annulus shield. See Figure 7.1.13.

24. Prepare for MPC lid welding as follows:

Note:
The following steps use two identical Removable Valve Operating Assemblies (RVOAs) (See Figure 7.1.21) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during vacuum drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

Note:
The vent and drain ports are opened by pushing the RVOA handle down to engage the square nut on the cap and turning the handle fully in the counter-clockwise direction. The handle will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the handle fully in the clockwise direction. The ports are closed when the handle cannot be turned further.

a. Clean the vent and drain ports to remove any dirt. Install the RVOAs (See Figure 7.1.21) to the vent and drain ports leaving caps open.

ALARA Warning:
Personnel should remain clear of the drain lines any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

b. Attach the water pump to the drain port (See Figure 7.1.22) and pump ~~approximately~~ *between 50 and* 120 gallons to the spent fuel pool or liquid radwaste system. The water level is lowered to keep moisture away from the weld region.

c. Disconnect the water pump.

25. Weld the MPC lid as follows:

ALARA Warning:
Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

Note:
The vacuum source *may* helps improve the weld quality by keeping moist air from condensing on the MPC lid weld area. The vacuum source can be supplied from a wet/dry vacuum cleaner or small vacuum pump.

a. Attach a vacuum source to the vent port *(if used)or inert the gas space under the MPC lid.*

ALARA Warning:
It may be necessary to rotate or reposition the MPC lid slightly to achieve uniform weld gap and lid alignment. A punch mark is located on the outer edge of the MPC lid and shell. These marks are aligned with the alignment mark on the top edge of the HI-STAR 100 overpack (See Figure 7.1.8). If necessary, the MPC lid lift should be performed using a hand operated chain fall to closely control the lift to allow rotation and repositioning by hand. If the chain fall is hung from the crane hook, the crane should be tagged out of service to prevent inadvertent use during this operation. Continuous radiation monitoring is recommended.

b. If necessary center the lid in the MPC shell using a hand-operated chain fall.

Note:
The MPC is equipped with lid shims that serve to close the gap in the joint for ~~MPC~~*MPC* lid closure weld.

c. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

ALARA Note:
The optional AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

d. Install the Automated Welding System baseplate shield *(if used)*. See Figure 7.1.12 for rigging.

e. Install the Automated Welding System Robot *(if used)*. See Figure 7.1.12 for rigging.

f. Tack weld the MPC lid.

g. Visually inspect the tack welds.

h. Lay the root weld.

Note:

The Lid-to-Shell weld may be examined by either volumetric examination (UT) or multi-layer liquid penetrant examination. If volumetric examination is used, it shall be the ultrasonic method and shall include a liquid penetrant (PT) of the root and final weld layers. If PT alone is used, at a minimum, it must include the root and final weld layers and one intermediate PT after approximately every 3/8 inch weld depth.

For all liquid penetrant examinations in this procedure, ASME Boiler and Pressure Vessel Code, Section V, Article 6 provides the liquid penetrant examination methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section III and V of the Code or site-specific program.

Volumetric examination of the MPC Lid-to-Shell weld by ultrasonic test methods are defined in ASME Boiler and Pressure Vessel Code, Section V, Article 5 and 2 respectively. The acceptance standards for UT examination are per Section III, Subsection NB, Article NB-5332 for UT as defined on the Design Drawings. NDE personnel shall be qualified per the requirements of Section III and V of the Code or site-specific program.

i. Disconnect the vacuum source from the vent port *(if used).*

j. Perform a liquid penetrant examination of the weld root.

k. Complete the MPC lid welding, performing at least one intermediate layer liquid penetrant examination after approximately *every* 3/8 inch weld depth.

l. Perform a liquid penetrant examination on the MPC lid final pass and UT (if required).

26. Perform hydrostatic and MPC leakage rate testing as follows:

ALARA Note:
The leakage rates are determined before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 7.1.23 for the hydrostatic test arrangement.

ALARA Warning:
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

b. Fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose.

c. Perform a hydrostatic test of the MPC as follows:

1. Close the drain valve and pressurize the MPC to 125 +5/-0 psig.

2. Close the inlet valve and monitor the pressure for a minimum of 10 minutes. The pressure shall not drop during the performance of the test.

3. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water. The acceptance criteria is no observable water leakage.

d. Release the MPC internal pressure, disconnect the water fill line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.

1. Repeat Step 25.1

e. Attach a regulated helium supply (pressure set to 10+10/-0 psig) to the vent port and attach the drain line to the drain port as shown on Figure 7.1.24.

~~f.Reset the totalizer on the drain line.~~

~~g.~~*f.* Verify the correct pressure (pressure set to 10+10/-0 psig) on the helium supply and open the helium supply valve. Drain approximately *5 to 10* ~~twenty~~ gallons. ~~as measured by the totalizer.~~

~~h.~~*g.* Close the drain port valve and pressurize the MPC to 10+10/-0 psig helium.

~~i.~~*h.* Close the vent port.

Note:
The leakage detector may detect residual helium in the atmosphere. If the leakage tests detects a leak, the area should be flushed with nitrogen or compressed air and the location should be retested.

~~j.~~*i.* Perform a helium sniffer probe leakage rate test of the MPC lid-to shell weld in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [7.1.5]. The MPC Helium Leak Rate shall be ≤ 5.0E-6 std cc/sec (He) with a minimum test sensitivity less than 2.5E-6 std cc/sec (He).

~~k.~~*j.* Repair any weld defects in accordance with the site's approved weld repair procedures. Reperform the Ultrasonic, Liquid Penetrant, Hydrostatic and Helium Leakage tests if weld repair is performed.

27. Drain the MPC as follows:

~~Note:~~
~~It is necessary to completely fill the MPC with water to get an accurate measurement of the MPC internal free space.~~

~~a.Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 7.1.23.~~

~~ALARA Warning:~~
~~Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.~~

~~b.Attach the water fill line to the drain port (water pressure set to 15+5/-0 psig) and fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the drain line.~~

~~c.Disconnect the water fill and drain lines from the MPC leaving the vent port valve open to allow for thermal expansion of the MPC water.~~

ALARA Warning:
Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

~~d.~~*a.* Attach a regulated helium or nitrogen supply (pressure set to ~~1~~25+5/-0 psig) to the vent port.

~~e.~~*b.* Attach a drain line to the drain port shown on Figure 7.1.24.

~~f.Reset the totalizer on the drain line.~~

~~g.~~*c.* Verify the correct pressure (pressure set to ~~1~~25+5/-0) on the gas supply.

~~h.~~*d.* Open the gas supply valve and record the time at the start of MPC draindown.

Note:
An optional warming pad may be placed under the HI-STAR 100 Overpack to replace the heat lost during the evaporation process of vacuum drying. This may be used at the user's discretion for older and colder fuel assemblies to reduce vacuum drying times.

~~i.~~*e.* Start the warming pad, if used.

~~j.~~*f.* Blow the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve.

~~k.Record the volume of water (as measured on the totalizer) drained from the MPC.~~

~~l.~~*g.* Disconnect the gas supply line from the MPC.

~~m.~~*h.* Disconnect the drain line from the MPC.

28. Vacuum Dry the MPC as follows:

Note:
Vacuum drying is performed to remove moisture and oxidizing gasses from the MPC. This ensures a suitable environment for long-term storage of spent fuel assemblies and ensures that the MPC pressure remains within design limits. The vacuum drying process reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC. *The vacuum stages are intermediate steps and should be considered approximate values.*

a. Attach the Vacuum Drying System (VDS) to the vent and drain port RVOAs. See Figure 7.1.25.

Note:
The Vacuum Drying System may be configured with an optional fore-line condenser to increase vacuum pump efficiency. Water may need to be periodically drained. The volume of condensed water should be measured and added to the water volume measured during MPC draining.

b. Reduce the MPC pressure to approximately 100 torr and throttle the VDS suction valve to maintain this pressure for approximately 15 minutes.

c. Reduce the MPC pressure to approximately 70 torr and throttle the VDS suction valve to maintain this pressure for approximately 15 minutes.

d. Reduce the MPC pressure to approximately 50 torr and throttle the VDS suction valve to maintain this pressure for approximately 15 minutes.

e. Reduce the MPC pressure to approximately 30 torr and throttle the VDS suction valve to maintain this pressure for approximately 15 minutes.

Note:
The Vacuum Drying System pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

f. When the MPC pressure begins to drop (without any operator action), completely open the VDS suction valve and reduce the MPC pressure to below 3 torr.

g. Shut the VDS valves and verify a stable MPC pressure on the vacuum gage.

Note:
The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the Vacuum Drying System, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

h. Perform the MPC dryness verification test. The MPC cavity shall hold stable vacuum drying pressure of ≤ 3 torr for ≥ 30 minutes.

i. Close the vent and drain port valves.

j. Disconnect the VDS from the MPC.

k. Stop the warming pad, if used.

l. Close the drain port RVOA cap and remove the drain port RVOA.

29. Backfill the MPC as follows:

> **Note:**
> Helium backfill requires 99.995% (minimum) purity. ~~The MPC-24 Helium Backfill Density (L^*_{He}) shall be 0.1212 +0/-10% g-moles/l. The MPC-68 Helium Backfill Density (L^*_{He}) shall be 0.1218 +0/-10% g-moles/l~~

a. *Set the helium bottle regulator pressure to 70+5/-0 psig.*

b. *Purge the Helium Backfill System to remove oxygen from the lines.*

c. *Attach the Helium Backfill System (HBS) to the vent port as shown on Figure 8.1.23 and open the vent port.*

d. *Slowly open the helium supply valve while monitoring the pressure rise in the MPC.*

> ***Note:***
> *If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.*

e. *Carefully backfill the MPC to between 0 (atmospheric) and 30 psig.*

f. *Disconnect the HBS from the MPC.*

~~a.~~ *g.* *Close the vent port RVOA and disconnect the vent port RVOA.*

~~*Calculate and record the maximum and minimum Helium backfill loading M_{max} and M_{min} respectively (g-moles):*~~

$$\cancel{M_{max} = V_W (L^*_{He} - e_L)}$$

$$\cancel{M_{min} = V_W (0.90L^*_{He} + e_L)}$$

~~where:~~

~~M_{max} = maximum helium backfill (g-moles)~~
~~M_{min} = minimum helium backfill (g-moles)~~
~~V_W = free volume of the MPC as measured during MPC draining and vacuum drying (liters)~~

~~L^*_{He} = backfill loading per unit volume as listed above (g-moles/liter) for each MPC type~~

~~e_L = absolute error in helium loading (g-moles/liter)~~

~~b. Set the helium bottle regulator pressure to 70+5/-0 psig.~~

~~c. Purge the Helium Backfill System to remove oxygen from the lines.~~

~~d. Attach the Helium Backfill System (HBS) to the vent port as shown on Figure 7.1.26 and open the vent port.~~

~~e. Reset the mass flow meter on the helium supply line.~~

~~f. Verify the correct pressure (pressure set to 70+5/-0 psig) and slowly open the helium supply valve while monitoring the helium mass flow meter.~~

~~g. Throttle the helium supply to regulate the flow rate to between 40 and 50 standard liters per minute.~~

~~Note:~~
~~If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.~~

~~h. Backfill the MPC to between M_{max} and M_{min} as determined in Step 29.~~

~~i. Disconnect the HBS from the MPC.~~

~~j. Close the vent port RVOA and disconnect the vent port RVOA.~~

30. Weld the vent and drain port cover plates as follows:

    a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.

    b. Place the cover plate over the vent port recess.

    c. Raise the edge of the cover plate and insert the nozzle of the helium supply into the vent port recess to displace the oxygen with helium.

Note:
Helium gas is required to be injected into the port recesses to ensure that the leakage test is valid.

    d. Displace the air in the recess using the helium nozzle and immediately close the cover plate.

    e. Tack weld the cover plate.

f. Visually inspect the tack welds.

g. Weld the root pass on the vent port cover plate.

> **Note:**
> ASME Boiler and Pressure Vessel Code [7.1.6], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

h. Perform a liquid penetrant examination on the vent port cover plate root weld.

i. Complete the vent port cover plate welding.

> **Note:**
> ASME Boiler and Pressure Vessel Code [7.1.6], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

j. Perform a liquid penetrant examination on the vent port cover weld.

k. Repeat Steps 30.a through 30.j for the drain port cover plate.

31. Perform a leakage test of the MPC vent and drain port cover plates as follows:

> **Note:**
> The leakage detector may detect residual helium in the atmosphere from the helium injection process. If the leakage tests detects a leak, the area should be blown clear with compressed air or nitrogen and the location should be retested.

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

b. Perform a helium leakage rate test of vent and drain cover plate welds in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [7.1.5]. The MPC Helium Leak Rate shall be $\leq$ 5.0E-6 std cc/sec (He) with a minimum test sensitivity less than 2.5E-6 std cc/sec (He).

c. Repair any weld defects in accordance with the site's approved code weld repair procedures. Reperform the leakage test as required.

32. Weld the MPC closure ring as follows:

ALARA Note:
The closure ring is installed by hand. No tools are required. The closure ring may be provided as a complete ring or in two halves. In the case of the single ring, no radial connecting welds are needed.

a. Install and align the closure ring. See Figure 7.1.8.

b. Tack weld the closure ring to the MPC shell and the MPC lid.

c. Visually inspect the tack welds.

d. Lay the root weld between the closure ring and the MPC shell *if necessary*.

e. Lay the root weld between the closure ring and the MPC lid *if necessary*.

f. Lay the root weld connecting the two closure ring segments, if necessary.

Note:
ASME Boiler and Pressure Vessel Code [7.1.6], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

g. Perform a liquid penetrant examination on the closure ring root welds.

h. Complete the closure ring welding.

Note:
ASME Boiler and Pressure Vessel Code [7.1.6], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination are contained in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

i. Perform a liquid penetrant examination on the closure ring final weld.

j. Remove the Automated Welding System.

k. If necessary, remove the AWS baseplate shield. See Figure 7.1.12 for rigging.

7.1.6 <u>Preparation for Transport</u>

1. Remove the annulus shield and seal surface protector and store it in an approved plant storage location

ALARA Warning:
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.

2. Attach a drain line to the HI-STAR 100 overpack drain connector and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system (See Figure 7.1.17).

3. Install the overpack closure plate as follows:

   a. Remove any waterproof tape or bolt plugs used for contamination mitigation.

   b. Clean the closure plate seal seating surface and the HI-STAR 100 overpack seal seating surface and install new overpack closure plate mechanical seals.

   c. Remove the test port plug and store it in a site-approved location. Discard any used metallic seals.

Note:
Care should be taken to protect the seal seating surface from scratches, nicks or dents.

d. Install the closure plate (see Figure 7.1.12). Disconnect the closure plate lifting eyes and install the bolt hole plugs in the empty bolt holes (See Table 7.1.3 for torque requirements).

e. Install and torque the closure plate bolts. See Table 7.1.3 for torque requirements.

f. Remove the vent port cover plate and remove the port plug and seal. Discard any used mechanical seals.

4. Dry the overpack annulus as follows:

a. Disconnect the drain connector from the overpack.

b. Install the drain port plug with a new seal and torque the plug. See Table 7.1.3 for torque requirements. Discard any used metallic seals.

Note:
Preliminary annulus vacuum drying may be performed using the test cover to improve flow rates and reduce vacuum drying time. Dryness testing and helium backfill shall use the backfill tool.

c. Load the backfill tool with the HI-STAR 100 overpack vent port plug and the vent port with a new plug seal. Attach the backfill tool to the HI-STAR 100 overpack vent port with the plug removed. See Figure 7.1.28. See Table 7.1.3 for torque requirements.

d. Evacuate the HI-STAR 100 overpack pressure to approximately 100 torr and hold the pressure for approximately 15 minutes.

e. Evacuate the HI-STAR 100 overpack pressure to approximately 50 torr and hold the pressure for approximately 15 minutes.

f. Evacuate the HI-STAR 100 overpack pressure to approximately 30 torr and hold the pressure for approximately 15 minutes.

g. Throttle the VDS suction valves to maintain about 30 torr and hold the pressure for approximately 15 minutes.

Note:
The Vacuum Drying System pressure will remain at about 30 torr until most of the liquid water has been removed from the overpack.

h. Continue to operate the Vacuum Drying System at about 30 torr while monitoring the HI-STAR 100 overpack pressure.

i. When the HI-STAR 100 overpack pressure begins to drop (without any operator action), completely open the Vacuum Drying System suction valve and reduce the HI-STAR 100 overpack pressure to below 3 torr.

Note:
The annulus pressure may rise due to the presence of water in the HI-STAR 100 overpack. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the Vacuum Drying System, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

j. Perform a HI-STAR 100 overpack Annulus Dryness Verification. The overpack annulus shall hold stable vacuum drying pressure of $\leq$ 3 torr for $\geq$ 30 minutes.

5. If necessary, perform a leakage test of the MPC-68F as follows:

a. Evacuate the annulus per the MSLD manufacturer's instructions and isolate the vacuum pump from the overpack test cover.

b. Perform a leakage rate test of MPC-68F per the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be $\leq$ 5.0E-6 std cc/sec (He) with a minimum test sensitivity less than 2.5E-6 std cc/sec (He).

6. Backfill, and leakage test the overpack as follows:

a. Attach the helium supply to the backfill tool.

b. Verify the correct pressure on the helium supply (pressure set to10+4/-0 psig) and open the helium supply valve.

c. Backfill the HI-STAR 100 overpack annulus to $\geq$ 10 psig and $\leq$ 14 psig.

d. Install the overpack vent port plug and torque. See Table 7.1.3 for torque requirements.

e. Disconnect the overpack backfill tool from the vent port.

f. Flush the overpack vent port recess with compressed air to remove any standing helium gas.

g. Install the overpack test cover to the overpack vent port as shown on Figure 7.1.27. See Table 7.1.3 for torque requirements.

h. Evacuate the test cavity per the MSLD manufacturer's instructions and isolate the vacuum pump from the overpack test cover.

i. Perform a leakage rate test of overpack vent port plug per the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be ≤ 4.3E-6 std cc/sec (He) with a minimum test sensitivity less than 2.15E-6 std cc/sec (He).

j. Remove the overpack test cover and install a new metallic seal on the overpack vent port cover plate. Discard any used metallic seals.

k. Install the vent port cover plate and torque the bolts. See Table 7.1.3 for torque requirements.

l. Repeat Steps 6.f through 6.k for the overpack drain port.

7. Leak test the overpack closure plate inner mechanical seal as follows:

   a. Attach the closure plate test tool to the closure plate test port with the and MSLD attached. See Figure 7.1.29. See Table 7.1.3 for torque requirements.

   b. Evacuate the closure plate test port tool and closure plate inter-seal area per the MSLD manufacturer's instructions.

   c. Perform a leakage rate test of overpack closure plate inner mechanical seal in accordance with the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be ≤ 4.3E-6 std cc/sec (He) with a minimum test sensitivity less than 2.15E-6 std cc/sec (He).

   d. Remove the closure plate test tool from the test port and install the test port plug with a new mechanical seal. See Table 7.1.3 for torque requirements. Discard any used metallic seals.

8. Drain the Temporary Shield Ring (Figure 7.1.20), if used. Remove the ring segments and store them in an approved plant storage location.

ALARA Warning:
For ALARA reasons, decontamination of the overpack bottom shall be performed using pole-mounted cleaning tools or other remote leaning devices.

ALARA Warning:
If the overpack is to be downended on the transport frame, the bottom shield should be installed quickly. Personnel should remain clear of the bottom of the unshielded overpack.

~~9.Verify that the HI-STAR 100 overpack dose rates are within the acceptance requirements listed above.~~

7.1.7 Placement of the HI-STAR 100 Overpack on the Transport Frame

1. Position the transport frame under the overhead lifting device.

2. Install the HI-STAR 100 overpack buttress plate on the HI-STAR 100 overpack. See Figure 7.1.11 and 7.1.12 for rigging. See Table 7.1.3 for torque requirements.

3. Downend the HI-STAR 100 overpack in the transport frame. See Section 7.1.2.

4. Install the removable shear ring segments. See Table 7.1.3 for torque requirements.

5. Perform a final inspection of the HI-STAR 100 overpack as follows:

ALARA Warning:
Dose rates around the unshielded bottom end of the HI-STAR 100 overpack may be higher that other locations around the overpack. Workers should exercise appropriate ALARA controls when working around the bottom end of the HI-STAR 100 overpack.

Note:
Prior to shipment of the HI-STAR 100 package, the accessible external surfaces of the HI-STAR 100 packaging (HI-STAR 100 overpack, impact limiters, personnel barrier, tie-down, transport frame and transport vehicle) shall be surveyed for removable radiological contamination and show less than 2200 dpm/100 cm^2 from beta and gamma emitting sources, and 220 dpm/100 cm^2 from alpha emitting sources.

   a. Perform a final decontamination of the HI-STAR 100 overpack, and survey for removable contamination.

   b. Perform a visual inspection of the HI-STAR 100 overpack to verify that there are no outward visual indications of impaired physical condition. Identify any significant indications to the cognizant individual for evaluation and resolution and record on the shipping documentation.

   c. Verify that the HI-STAR 100 overpack neutron shield rupture discs are installed, intact and not covered by tape or other covering.

6. Install the tie-down over the HI-STAR 100 overpack and secure the tie-down bolts. See Table 7.1.3 for torque requirements.

7. Install the impact limiters as follows:

   a. Install the alignment pins in the bottom of the HI-STAR 100 overpack. See Figure 7.1.11. See Table 7.1.3 for torque requirements.

   b. Using the impact limiter handling frame, raise and position the impact limiter over the end of HI-STAR 100. See Figure 7.1.10.

c. Install the impact limiter bolts. See Table 7.1.3 for torque requirements.

d. Repeat for the other impact limiter.

Note:
The impact limiters cover all the HI-STAR 100 penetrations. The security seals are used to provide tamper detection.

e. Install a security seal (one each) through the threaded hole in the top and bottom impact limiter bolts. Record the security seal number on the shipping documentation.

f. Perform final radiation surveys of the package surfaces per 10CFR71.47 [7.0.1] and SAR Section 8.1.5.2~~.1~~ and 49CFR173.443 [7.1.3]. Record the results on the shipping documentation.

8. Install the personnel barrier as follows:

a. Rig the personnel barrier as shown in Figure 7.1.9 and position the personnel barrier over the frame.

b. Remove the personnel barrier rigging and install the personnel barrier locks.

c. Transfer the personnel barrier keys to the carrier.

9. Perform a final check to ensure that the package is ready for release as follows:

a. Verify that required radiation survey results are properly documented on the shipping documentation.

b. Perform a HI-STAR 100 overpack surface temperature check. The accessible surfaces of the HI-STAR 100 Package (impact limiters and personnel barrier) shall not exceed 185 °F when measured in the shade in still air.

c. Verify that all required leakage testing has been performed and the acceptance criteria has been met and document the results on the shipping documentation.

d. Verify that the receiver has been notified of the impending shipment and that the receiver has the appropriate procedures and equipment is available to safely receive and handle the HI-STAR 100 System (10CFR20.1906(e)) [7.1.4].

e. Verify that the carrier has the written instructions and a list of appropriate contacts for notification of accidents or delays.

f. Verify that the carrier has written instructions that the shipment is to be Exclusive Use in accordance with 49CFR172.443 [7.1.3].

g. Verify that route approvals and notification to appropriate agencies have been completed.

h. Verify that the appropriate labels have been applied in accordance with 49CFR172.403.

i. Verify that the appropriate placards have been applied in accordance with 49CFR172.500.

j. Verify that all required information is recorded on the shipping documentation.

10. Release the HI-STAR 100 System for transport.

Table 7.1.1

HI-STAR 100 SYSTEM COMPONENT AND HANDLING WEIGHTS

Component	Weight (lbs)		Case Applicability†			
	MPC-24	MPC-68	1	2	3	4
Empty HI-STAR Overpack (without Closure Plate)	145,726	145,726	1	1	1	1
HI-STAR Closure Plate (without rigging)	7,984	7,984		1	1	1
Empty MPC (without Lid or Closure Ring)	29,075	*28,502*~~28,679~~	1	1	1	1
MPC Lid (without Fuel Spacers or Drain Line)	9677	10,194	1	1	1	1
MPC Closure Ring	145	145		1	1	1
MPC Lower Fuel Spacers††	401	258	1	1	1	1
MPC Upper Fuel Spacers††	144	315	1	1	1	1
MPC Drain Line	50	50	1	1	1	1
Fuel (Design Basis)	36,360	*42,092*~~42,160~~	1	1	1	1
Non Fuel Components	*3,960*~~0~~	5,440	~~1~~	~~1~~	~~1~~	~~1~~
Damaged Fuel Container (Dresden 1)	0	150				
Damaged Fuel Container (Humboldt Bay)	0	120				
MPC Water ††† (with Fuel in MPC)	*17,630*~~16,433~~	*16,957*~~16,163~~	1			
Annulus Water	280	280	1			
HI-STAR Lift Yoke (with slings)	3600	3600	1	1		
Annulus Seal	50	50	1			
Lid Retention System	2300	2300				
Transport Frame	9000	9000			1	
Temporary Shield Ring	2500	2500				
Automated Welding System Baseplate Shield	2000	2000				
Automated Welding System Robot	1900	1900				
Top Impact Limiter (without Buttress Plate)	16,667	16,667			1	1
Bottom Impact Limiter	17,231	17,231			1	1
Impact Limiter Handling Frame	1980	1980				
Buttress Plate	2520	2520			1	1
Tie-Down	995	995			1	
Personnel Barrier	1500	1500			1	

† See Table 7.1.2.

†† The fuel spacers referenced in this table are for the heaviest fuel assembly for each MPC. This yields the maximum weight of fuel assemblies and spacers.

††† Varies by fuel type and loading configuration. Users may opt to pump some water from the MPC prior to removal from the spent fuel pool to reduce the overall lifted weight.

TABLE 7.1.2
MAXIMUM HANDLING WEIGHTS
HI-STAR 100 SYSTEM

Caution:
The maximum weight supported by the HI-STAR 100 overpack lifting trunnions (not including the lift yoke) cannot exceed 250,000 lbs. Users should determine their specific handling weights based on the MPC contents and the expected handling modes.

Note:
The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations.

Case No.	Load Handling Evolution	Weight (lbs)	
		MPC-24	MPC-68
1	Loaded HI-STAR Removal from Spent Fuel Pool	*242,993*~~241,796~~	*248,024*~~252,915~~
2	Loaded HI-STAR During Movement through Hatchway	*233,162*~~233,162~~	*238,866*~~244,551~~
3	Weight on Transport Vehicle	*277,475*~~277,475~~	*283,179*~~288,864~~
4	Gross HI-STAR 100 Package Weight	*265,980*~~265,980~~	*271,684*~~277,369~~

Table 7.1.3
HI-STAR 100 SYSTEM TORQUE REQUIREMENTS

Fastener	Torque (ft-lbs)	Pattern
Overpack Closure Plate Bolts†, ††	First Pass – Hand Tight Second Pass – Wrench Tight Third Pass – 860+25/-25 Fourth Pass – 1725+50/-50 Final Pass - 2895+90/-90	Figure 7.1.30
Overpack Vent and Drain Port Cover Plate Bolts††	12+2/-0	X-pattern
Overpack Vent and Drain Port Plugs	~~22+2/-0~~ *45+5/-0*	None
Closure Plate Test Port Plug	22+2/-0	None
Backfill Tool Test Cover Bolts††	16+2/-0	X-pattern
Shear Ring Segments	22+2/-0	None
Overpack Bottom Cover Bolts	200+20/-0	None
Pocket Trunnion Plugs	Hand Tight	None
Threaded Fuel Spacers	Hand Tight	None
MPC Lid Threaded Plugs	Hand Tight	None
Impact Limiter Alignment Pin	Hand Tight	None
Top Impact Limiter Attachment Bolt	256+10/-0	None
Bottom Impact Limiter Attachment Bolt	1500+45/-0	None
Buttress Plate Bolts	150+10/-0	None
Tie-Down Bolts	250+20/-0	None
Transport Frame Bolts	250+20/-0	None

† Detorquing shall be performed by turning the bolts counter-clockwise in 1/3 turn +/- 30 degrees increments per pass according to Figure 7.1.30 for three passes. The bolts may then be removed.

†† Bolts shall be cleaned and inspected for damage or excessive wear (replaced if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

Table 7.1.4
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure	Description
Annulus Overpressure System (optional)	Not Important To Safety	7.1.17	The Annulus Overpressure System is used for supplemental protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure. The Annulus Overpressure System consists of the quick disconnects water reservoir, reservoir valve and annulus connector hoses. User is responsible for supplying demineralized water to the location of the Annulus Overpressure System.
Annulus Shield (optional)	Not Important To Safety	7.1.13	A segmented solid shield that is placed at the top of the annulus to provide supplemental shielding to the operators performing cask loading and closure operations. Shield segments are installed by hand, no crane or tools required.
Automated Welding System (optional)	Not Important To Safety	7.1.2b	Used for remote welding of the MPC lid, vent and drain port cover plates and the MPC closure ring. The AWS consists of the robot, wire feed system, torch system, weld power supply and gas lines.
AWS Baseplate Shield (optional)	Not Important To Safety	7.1.2b	The AWS baseplate shield provides supplemental shielding to the operators during the cask closure operations.
Backfill Tool	Not Important to Safety	7.1.28	Used to dry, backfill the HI-STAR 100 annulus and install the HI-STAR 100 overpack vent and drain port plugs. The backfill tool uses the same bolts as the HI-STAR 100 overpack vent and drain cover plates.
Blowdown Supply System	Not Important To Safety	7.1.24	Gas hose with pressure gauge, regulator used for blowdown of the MPC.
Closure Plate Test Tool	Not Important to Safety	7.1.29	Used to helium leakage test the HI-STAR 100 overpack Closure Plate inner mechanical seal.
Cool-Down System	Not Important To Safety	7.2.5	The Cool-Down System is a closed-loop forced ventilation cooling system used to gas-cool the MPC fuel assemblies down to a temperature water can be introduced without the risk of thermally shocking the fuel assemblies or flashing the water, causing uncontrolled pressure transients. The Cool-Down System is attached between the MPC drain and vent ports The CDS consists of the piping, blower, heat exchanger, valves, instrumentation, and connectors. The CDS is used only for unloading operations.
Drain Connector	Not Important To Safety	7.1.17	Used for draining the annulus water following cask closure operations. The Drain Connector consists of the connector pipe valve, and quick disconnect for adapting to the Annulus Overpressure System.

Table 7.1.4 (Continued)
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure	Description
Four Legged Sling and Lifting Rings	Not Important To Safety (controlled under the user's rigging equipment program)	7.1.12	Used for rigging the HI-STAR 100 overpack upper shield lid, MPC lid, Automated Welding System Baseplate shield, and Automated Welding System Baseplate Shield. Consists of a four legged sling, lifting rings, shackles and a main lift link.
Helium Backfill System	Not Important To Safety	7.1.26	Used for helium backfilling of the MPC. System consists of the gas lines, mass flow monitor, integrator, and valved quick disconnect.
Hydrostatic Test System	Not Important to Safety	7.1.23	Used to hydrostatically test the MPC primary welds. The hydrostatic test system consists of the gauges, piping, pressure protection system piping and connectors.
Impact Limiter Handling Frame	Not Important to Safety	7.1.10	The impact limiter handling frame is used for installing, removing, handling and storing the impact limiters. The impact limiter handling frame consists of the handling frame and rigging.
Impact Limiters	Important to Safety	7.1.11	The impact limiters are used to limit the HI-STAR 100 decelerations to less than 60 g during postulated transportation accidents. The impact limiters consist of the top and bottom impact limiter and the connecting fasteners.
Inflatable Annulus Seal	Not Important To Safety	7.1.13	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System (optional)	User designated	7.1.18	The Lid Retention System provides three functions; it guides the MPC lid into place during underwater installation, establishes lift yoke alignment with the HI-STAR 100 overpack trunnions, and locks the MPC lid in place during cask handling operations between the pool and decontamination pad. The device consists of the retention disk, alignment pins, lift yoke connector links and lift yoke attachment bolts.
Lift Yoke	User designated	7.1.3	Used for HI-STAR 100 overpack cask handling when used in conjunction with the overhead crane. The lift yoke consists of the lift yoke assembly and crane hook engagement pin(s). The lift yoke is a modular design that allows inspection, disassembly, maintenance and replacement of components.
MPC Fill Pump System (optional)	Not Important To Safety	Not shown	Large pump used for filling the MPC with spent fuel pool water prior to cask insertion into the spent fuel pool. Also used for emptying of the MPC for unloading operations.
MPC Upending Frame	Not Important to Safety	7.1.6	A welded steel frame used to evenly support the MPC during upending operations. The frame consists of the main frame, MPC support saddles, two rigging bars, wrap around-straps, and strap attachment lugs.

Table 7.1.4 (Continued)
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure	Description
MSLD (Helium Leakage Detector)	Not Important To Safety	Not shown	Used for helium leakage testing of the MPC closure welds.
MSLD Calibration Sources	Not Important To Safety	Not shown	Traceable leakage sources for periodic calibration of the MSLD.
Overpack Bottom Cover (optional)	Not Important to Safety	Not shown	A cup-shaped shield used to reduce dose rates around the HI-STAR 100 overpack bottom end when operated in the horizontal orientation.
Overpack Test Cover	Not Important to Safety	7.1.27	Used to helium leakage test the HI-STAR 100 overpack vent and drain port plug seals.
Personnel Barrier	Not Important to Safety	7.1.9	The personnel barrier is a ventilated enclosure cage that fits over the main body of the HI-STAR 100 overpack. The personnel barrier is designed to restrict personnel accessibility to the potentially hot surfaces of the HI-STAR 100 overpack. The personnel barrier in conjunction with the impact limiters restrict accessibility to all surfaces of the HI-STAR 100 overpack during transport. The personnel barrier is equipped with locks to prevent unauthorized access. The personnel barrier is equipped with a four-legged bridle sling used for installation and removal.
Seal Surface Protector (optional)	Not Important to Safety	7.1.13	Used to protect the HI-STAR 100 overpack mechanical seal seating surface during loading and MPC closure operations.
Small Water Pump (optional)	Not Important To Safety	7.1.22	Used for lowering the MPC water level prior to lid welding. The small water pump consists of the pump, hose and connector fittings.
Temporary Shield Ring (optional)	Not Important To Safety	7.1.20	A water-filled segmented tank that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations. Shield segments are installed by hand, no tools are required.
Threaded Inserts	Not Important To Safety	Not shown	Used to fill the empty threaded holes in the HI-STAR 100 overpack and MPC.
Tie-Down	Not Important to Safety	7.1.11	The tie-down is a horse-shoe shaped collar that secures the HI-STAR 100 top end to the transport frame. The tie-down is secured by multiple bolts.
Transport Frame	Not Important To Safety	7.1.6	A welded steel frame used to support the HI-STAR 100 overpack during on-site movement and upending/downending operations. The frame consists of the rotation trunnions, main frame beams and front saddle.

Table 7.1.4 (Continued)
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure	Description
Transport Vehicle	Not Important to Safety	Not Shown	Any flatbed rail car, heavy haul trailer or other vice used to transport the loaded HI-STAR 100 overpack.
Vacuum Drying System	Not Important To Safety	7.1.25	Used for removal of residual moisture from the MPC and HI-STAR 100 Overpack annulus following water draining. Used for evacuation of the MPC to support backfilling operations. Used to support test volume samples for MPC unloading operations. The VDS consists of the vacuum pump, piping, skid, gauges, valves, inlet filter, flexible hoses, connectors, control system.
Vacuum Drying System Fore-Line Condenser (optional)	Not Important to Safety	Not Shown	Optional item used to improve the Vacuum Drying System pump efficiency. The condenser removes water from the vacuum stream prior to the vacuum pump.
Vent and Drain RVOAs (optional)	Not Important To Safety	7.1.21	Used to drain, dry, inert and fill the MPC through the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Warming Pad (optional)	Not Important to Safety	Not Shown	Used to improve vacuum drying time for older and colder fuel assemblies. The pad consists of the heater pad, heater, circulation pump, expansion tank, hoses and fittings. Other configurations are acceptable.
Water Totalizers	Not Important To Safety	7.1.22 and 7.1.24	Used for water pump-down prior to lid welding operations *and water removal for MPC helium leakage testing*. ~~Used for determining the MPC free space volume which is the input for the calculation of the required helium backfill quantities.~~
Weld Removal System (optional)	Not Important To Safety	7.2.2b	Semi-automated weld removal system used for removal of the MPC to shell weld, MPC to closure ring weld and closure ring to MPC shell weld. The WRS mechanically removes the welds using a high-speed cutter.

Table 7.1.5
HI-STAR 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Note:
The following list summarizes the instruments identified in the procedures for cask loading and unloading operations. Alternate instruments are acceptable as long as they can perform appropriate measurements.

Instrument	**Function**
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors the air flow rate during assembly cool-down.
Helium Mass Flow Monitor	Determines the amount of helium introduced into the MPC during backfilling operations. Includes integrator.
Helium Mass Spectrometer Leak Detector (MSLD)	Ensures leakage rates of welds are within acceptance criteria.
Helium Pressure Gauges	Ensures correct helium backfill pressure during backfilling operation.
Volumetric Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauge	Ensures correct helium pressure during fuel cool-down operations.
Hydrostatic Test Pressure Gauge	Used for hydrostatic testing of MPC lid-to-shell weld.
Temperature Gauge	Monitors the state of fuel cool-down prior to MPC flooding.
Temperature Probe	For fuel cool-down operations
Vacuum Gauges	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Water Pressure Gauge	Used for performance of the ~~MPC~~*MPC* Hydrostatic Test.
Water Totalizer	Used for water pump-down prior to lid welding operations *and water removal for MPC helium leakage testing*. ~~Used for determining the MPC free space volume which is the input for the calculation of the required helium backfill quantities.~~

† All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 7.1.6
HI-STAR 100 ~~SYSTEM~~ *OVERPACK* SAMPLE INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STAR 100 overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

<u>HI-STAR 100 Overpack Closure Plate:</u>

1. Lifting rings shall be inspected for general condition and date of required load test certification.
2. The test port shall be inspected for dirt and debris, hole blockage, thread condition, presence or availability of the port plug and replacement mechanical seals.
3. The mechanical seal grooves shall be inspected for cleanliness, dents, scratches and gouges and the presence or availability of replacement mechanical seals.
4. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
5. All closure plate surfaces shall be relatively free of dents, scratches, gouges or other damage.
6. The vent port plug shall be inspected for thread condition, and sealing surface condition (scratches, gouges).
7. Overpack vent port shall be inspected for presence or availability of port plugs, hole blockage, plug seal seating surface condition.
8. Overpack vent port cover plate shall be inspected for cleanliness, scratches, dents, and gouges, availability of retention bolts, availability of replacement mechanical seals.

<u>HI-STAR 100 Overpack Main Body:</u>

1. The impact limiter attachment bolt holes shall be inspected for dirt and debris and thread condition.
2. The mechanical seal seating surface shall be inspected for cleanliness, scratches, and dents or gouges.
3. The drain port plug shall be inspected for thread condition, and sealing surface condition (scratches, gouges).
4. The closure plate bolt holes shall be inspected for dirt, debris and thread damage.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Trunnions shall be inspected for deformation, cracks, thread damage, end plate damage, corrosion, excessive galling, damage to the locking plate, presence or availability of locking plate and end plate retention bolts.

Table 7.1.6
HI-STAR 100 OVERPACK *SAMPLE* INSPECTION CHECKLIST
(continued)

~~1.~~*7.* Pocket trunnion recesses shall be inspected for indications of over stressing (i.e., cracks, deformation, excessive wear).
~~2.~~*8.* Overpack drain port cover plate shall be inspected for cleanliness, scratches, dents, and gouges, availability of retention bolts, availability of replacement mechanical seals.
~~3.~~*9.* Overpack drain port shall be inspected for presence or availability of port plug, availability of replacement mechanical seals, hole blockage, plug seal seating surface condition.
~~4.~~*10.* Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
~~5.~~*11.* The overpack rupture disks shall be inspected for presence or availability and the top surface of the disk shall be visually inspected for holes, cracks, tears or breakage.
~~6.~~*12.* The nameplate shall be inspected for presence and general condition.
~~7.~~*13.* The removable shear ring shall be inspected for fit and thread condition.

Table 7.1.7
MPC SAMPLE INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:
1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
4. Lower fuel spacers (if used) shall be inspected for availability and general condition.
5. Drain and vent port cover plates shall be inspected for availability and general condition.
6. Serial numbers shall be inspected for readability.

MPC Main Body:
1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.

7.2 PROCEDURE FOR UNLOADING THE HI-STAR 100 SYSTEM IN THE SPENT FUEL POOL

7.2.1 Overview of HI-STAR 100 System Unloading Operations

ALARA Note:
The procedure described below uses the Weld Removal System, a remotely operated system that mechanically removes the welds. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STAR 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-STAR 100 overpack and empty MPC. These procedures are only used to respond to extreme abnormal events. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over-pressurization and thermal shock to the stored spent fuel assemblies. Figure 7.2.1 shows a flow diagram of the HI-STAR 100 overpack unloading operations. Figure 7.2.2 illustrates the major HI-STAR 100 overpack unloading operations.

Refer to the boxes of Figure 7.2.2 for the following description. The HI-STAR 100 overpack is received from the carrier, inspected and surveyed. The personnel barrier is removed and the impact limiters and tie-down are removed. The HI-STAR 100 overpack is upended and returned to the fuel building (Box 1). The HI-STAR 100 overpack vent port cover plate is removed and a gas sample is drawn from the HI-STAR 100 overpack annulus to determine the condition of the MPC confinement boundary. The annulus is depressurized and the HI-STAR 100 overpack closure plate is removed (Box 2). The Temporary Shield Ring is installed on the HI-STAR 100 overpack upper section. The Temporary Shield Ring and annulus are filled with plant demineralized water. The annulus shield is installed to protect the annulus from debris produced from the lid removal process. The MPC closure ring *above the vent and drain ports and the vent and drain port cover plates are core-drilled and removed to access the vent and drain ports.* ~~weld is removed using the Weld Removal System. The closure ring is removed. The vent port cover plate weld is removed and the cover plate is removed~~ (Box 3). The design of the vent and drain ports use metal-to-metal seals that prevent rapid decompression of the MPC and subsequent spread of contamination during unloading. The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. ~~A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The drain port cover plate weld is removed and the cover plate is removed.~~

The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding (Box 4). The cool-down process gradually reduces the cladding temperature to a point where the MPC may be flooded with water without thermally shocking the fuel assemblies or over-pressurizing the MPC from the formation of steam. Following the fuel cool-down, the MPC is filled with water at a specified rate (Box 5). The Weld Removal System then removes the MPC lid to MPC shell weld. The Weld Removal System is removed with the MPC lid left in place (Box 6).

The top surfaces of the HI-STAR 100 overpack and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or lid retention system and the lift yoke is engaged to the HI-STAR 100 overpack lifting trunnions. The HI-STAR 100 overpack is placed in the spent fuel pool and the MPC lid is removed (Box 7). All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris and crud (Box 8). The HI-STAR 100 overpack and MPC are returned to the designated preparation area (Box 9) where the MPC water is pumped back into the spent fuel pool, liquid radwaste system or other approved location. The annulus water is drained and the MPC and overpack are decontaminated (Box 10 and 11).

7.2.2 HI-STAR 100 Overpack Packaging Receipt

1. Recover the shipping documentation from the carrier along with the keys to the personnel barrier locks.

2. Remove the personnel barrier (See Figure 7.1.9) and perform a partial visual inspection of the HI-STAR 100 overpack to verify that there are no outward visual indications of impaired physical conditions. Identify any significant indications to the cognizant individual for evaluation and resolution.

ALARA Warning:
Dose rates around the unshielded bottom end of the HI-STAR 100 overpack may be higher that other locations around the overpack. Workers should exercise appropriate ALARA controls when working around the bottom end of the HI-STAR 100 overpack.

3. Remove the impact limiters as follows:

   a. Record the impact limiter security seal serial numbers and verify that they match the corresponding shipping documentation, as applicable.

   b. Clip the security seal wires and remove the security seals and wires.

   c. Attach the impact limiter handling frame as shown on Figure 7.1.10.

Caution:
The slings should be preloaded to the impact limiter weight plus the weight of the impact limiter handling frame prior to removal of the impact limiter bolts. (See Table 7.1.1) This will prevent movement of the impact limiter and damage to the bolts from excessive lift pressure during bolt removal.

   d. Using the load cell, apply the correct lift load. See Table 7.1.1.

e. Remove the bolts securing the impact limiter to the overpack. See Figure 7.1.11.

f. Remove the impact limiter and store the impact limiter in a site-approved location.

g. Repeat Steps 3.c. through 3.f. for the other impact limiter.

4. Complete the HI-STAR 100 overpack visual inspection to verify that there are no outward visual indications of impaired physical conditions. Identify any significant indications to the cognizant individual for evaluation and resolution.

Note:
On receipt of the loaded or empty HI-STAR 100 packaging, the accessible external surfaces of the HI-STAR 100 packaging (HI-STAR 100 overpack, impact limiters, personnel barrier, tie-down, transport frame and transport vehicle) shall be surveyed for removable radiological contamination and show less than 2200 dpm/100 cm^2 from beta and gamma emitting sources, and 220 dpm/100 cm^2 from alpha emitting sources.

5. Perform a radiation survey and removable contamination survey. Record the results on the shipping documentation.

6. Verify that the HI-STAR 100 overpack neutron shield rupture discs are installed and intact. Identify any non-conformances to the cognizant individual for evaluation and resolution.

7. If necessary, upend the HI-STAR 100 overpack in accordance with Section 7.1.2.

8. Transfer the HI-STAR 100 overpack to the fuel building.

9. Remove the buttress plate. See Figure 7.1.11 and 7.1.12.

10. Place the HI-STAR 100 overpack in the designated preparation area.

7.2.3 Preparation for Unloading

ALARA Warning:
Gas sampling is performed to assess the condition of the MPC confinement boundary. If a leak is discovered in the MPC boundary, the user's Radiation Control organization may require special actions to vent the HI-STAR 100.

1. Perform annulus gas sampling as follows:

a. Remove the overpack vent port cover plate and attach the backfill tool with a sample bottle attached. See Figure 7.2.3. Store the cover plate in a site-approved location.

b. Using a vacuum pump, evacuate the sample bottle and backfill tool.

c. Slowly open the vent port plug and gather a gas sample from the annulus. Reinstall the HI-STAR 100 overpack vent port plug.

d. Evaluate the gas sample and determine the condition of the MPC confinement boundary.

2. If the confinement boundary is intact (i.e., no radioactive gas is measured) then vent the overpack annulus by removing the overpack vent port seal plug (using the backfill tool). Otherwise vent the annulus gas in accordance with instructions from Radiation Protection.

3. Remove the closure plate bolts. See Table 7.1.3 for detorquing requirements. Store the closure plate bolts in a site-approved location.

4. Remove the overpack closure plate. See Figure 7.1.12 for rigging. Store the closure plate on cribbing to protect the seal seating surfaces.

5. Install the HI-STAR 100 overpack Seal Surface Protector (See Figure 7.1.13).

Warning:
Annulus fill water may flash to steam due to high MPC shell temperatures. Water addition should be performed in a slow and controlled manner.

6. Remove the HI-STAR 100 overpack drain port cover and port plug and install the drain connector. Store the drain port cover plate and port plug in an approved storage location.

7. Slowly fill the annulus area with plant demineralized water to approximately 4 inches below the top of the MPC shell and install the annulus shield. The annulus shield reduces the dose around the annulus area and prevents debris from entering the annulus during MPC lid weld removal operations. See Figure 7.1.13.

8. Remove the MPC closure *welds*~~ring~~ as follows:

ALARA Note:
The following procedures describe weld removal using the Weld Removal System. The Weld Removal System removes the welds with a high speed machine tool head. A vacuum head is attached to remove a majority of the metal shavings. Other methods of opening the MPC are acceptable.

ALARA Warning:
Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

a. Install bolt plugs and/or waterproof tape on the closure plate bolt holes.

b. *Install the Weld Removal System on the MPC lid and core drill through the closure ring and vent and drain port cover plate welds.*

~~b.Install the Weld Removal System on the MPC lid and remove the closure ring inner and outer welds.~~

9. *Access the vent and drain ports.*

~~c.~~*a.* ~~Remove the closure ring.~~

~~9.~~*10.* Remove the vent port cover plate weld and remove the vent port cover plate.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

~~10.~~*11.* Take an MPC gas sample as follows:

a. Attach the RVOA to the vent port (See Figure 7.1.21).

b. Attach a sample bottle to the vent port RVOA as shown on Figure 7.2.4.

c. Using the Vacuum Drying System, evacuate the RVOA and Sample Bottle.

d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

~~g.Remove the drain port cover plate weld and remove the cover plate.~~

~~h.~~*g.* Install the RVOA in the drain port.

~~11.~~*12.* Perform Fuel Assembly Cool-Down as follows:

a. Configure the Cool-Down System as shown on Figure 7.2.5.

b. Verify that the helium gas pressure regulator is set to *less than 100*~~85+3/-3~~ psig.

c. Open the helium gas supply valve to purge the gas lines of air.

Note:
The coolant flow direction is into the drain port and out of the vent port.

d. Confirm the heat exchanger coolant flow direction.

e. If necessary, slowly open the helium supply valve and increase the Cool-Down System pressure to *MPC pressure*~~100 +3/-3 psig~~. Close the helium supply valve.

f. Start the gas coolers.

g. Open the vent and drain port caps using the RVOAs.

h. Start the blower and monitor the gas exit temperature. Continue the fuel cool-down operations until the gas exit temperature is ≤ 200° F. These conditions shall be met prior to initiation of MPC re-flooding operations.

Note:
Water filling should commence immediately after the completion of fuel cool-down operations to minimize prevent fuel assembly heat-up. Prepare the water fill and vent lines in advance of water filling.

i. Prepare the MPC fill and vent lines as shown on Figure 7.1.23. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Turn off the blower and disconnect the gas lines to the vent and drain port RVOAs. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

j. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a ~~flow rate of between 12 to18 gallons per minute (gpm) and a~~ pressure less than 90 psi. Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

k. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.

l. Remove the *closure ring-to-MPC shell weld and the* MPC lid-to-shell weld using the Weld Removal System and remove the Weld Removal System. See Figure 7.1.12 for rigging.

m. Vacuum the top surfaces of the MPC and the HI-STAR 100 overpack to remove any metal shavings.

~~12.~~*13.* Install the inflatable annulus seal as follows:

Caution:
Do not use any sharp tools or instruments to install the inflatable seal.

a. Remove the annulus shield.

b. Manually insert the inflatable seal around the MPC. See Figure 7.1.13.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal between 30 and 35 psig *or as directed by the manufacturer.*

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.

~~13.~~*14.* Place HI-STAR 100 overpack in the spent fuel pool as follows:

a. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions, remove the MPC lid lifting threaded inserts and attach the MPC lid slings or Lid Retention System to the MPC lid.

b. If the Lid Retention System is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.

c. Install the Lid Retention System bolts if the Lid Retention System is used.

ALARA Note:
The Annulus Overpressure System is used to provide additional protection against MPC external shell contamination during in-pool operations. The Annulus Overpressure System is equipped with double locking quick disconnects to prevent inadvertent draining. The reservoir valve must be closed to ensure that the annulus is not inadvertently drained through the Annulus Overpressure System when the cask is raised above the level of the annulus reservoir.

d. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-STAR 100 overpack via the quick disconnect. See Figure 7.1.17.

Warning:
Cask placement in the spent fuel pool is the heaviest lift that occurs during the HI-STAR 100 unloading operations. The HI-STAR 100 trunnions must not be subjected to lifted loads in excess of 250,000 lbs. Users must ensure that plant-specific lifting equipment is qualified to lift the expected load. Users may elect to pump a measured quantity of water from the MPC prior to placement of the HI-STAR 100 in the spent fuel pool. See Table 7.1.1 and 7.1.2 for weight information.

e. Position the HI-STAR 100 overpack over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

f. Wet the surfaces of the HI-STAR 100 overpack and lift yoke with plant demineralized water while slowly lowering the HI-STAR 100 overpack into the spent fuel pool.

g. When the top of the HI-STAR 100 overpack reaches the approximate elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

h. If the Lid Retention System is used, remove the lid retention bolts when the top of the HI-STAR 100 overpack is accessible from the operating floor.

i. Place the HI-STAR 100 overpack on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

> **Note:**
> An underwater camera or other suitable viewing device may be used for monitoring the underwater operations.

j. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.

k. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.

> **Warning:**
> The MPC lid and unloaded MPC may contain residual contamination. All work done on the unloaded MPC should be carefully monitored and performed.

l. Disconnect the drain line from the MPC lid.

m. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.

n. Disconnect the Lid Retention System if used.

7.2.4 <u>MPC Unloading</u>

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Vacuum the cells of the MPC to remove any debris or corrosion products.
3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

7.2.5 <u>Post-Unloading Operations</u>

1. Remove the HI-STAR 100 overpack and the unloaded MPC from the spent fuel pool as follows:

   a. Engage the lift yoke to the top trunnions.

   b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.

   c. Raise the HI-STAR 100 overpack until the HI-STAR 100 overpack flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of the HI-STAR 100 overpack during fuel unloading.

   d. Measure the dose rates at the top of the HI-STAR 100 overpack in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

   e. Raise the top of the HI-STAR 100 overpack and MPC to the level of the spent fuel pool deck.

   f. Close the Annulus Overpressure System reservoir valve if the Annulus Overpressure System was used.

   g. Using a water pump, lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of the HI-STAR 100 overpack, the surfaces of the HI-STAR 100 overpack and lift yoke should be kept wet until decontamination can begin.

   h. Remove the HI-STAR 100 overpack from the spent fuel pool while spraying the surfaces with plant demineralized water.

   i. Disconnect the Annulus Overpressure System from the HI-STAR 100 overpack via the quick disconnect. Drain the Annulus Overpressure System lines and reservoir.

   j. Place the HI-STAR 100 overpack in the designated preparation area.

   k. Disengage the lift yoke.

   l. Perform decontamination on the HI-STAR 100 overpack and the lift yoke.

2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

4. Drain the water in the annulus.

5. Remove the MPC from the HI-STAR 100 overpack and decontaminate the MPC as necessary.

6. Decontaminate the HI-STAR 100 overpack.

7. Remove any bolt plugs, seal surface protector and/or waterproof tape from the HI-STAR 100 overpack top bolt holes.

8. Move the HI-STAR 100 overpack and MPC for further inspection, corrective actions, or disposal as necessary.

7.3 PREPARATION OF AN EMPTY PACKAGE FOR TRANSPORT

7.3.1 Overview of the HI-STAR 100 System Empty Package Transport

The operations for preparing *an*the empty package *(previously used)* for transport are similar to those required for transporting the loaded package with several exceptions. The closure plate is installed and the bolts are torqued. The HI-STAR 100 overpack is downended on the transport frame and the tie-down is installed. A survey for removable contamination is performed to verify that the removable contamination on the internal and external surfaces of the HI-STAR 100 overpack are ALARA and that the limits of 49CFR173.428 [7.1.3] and 10CFR71.87(i) [7.0.1] are met. At the User's discretion, impact limiters are installed and the personnel barrier is installed and locked. The procedures provided herein describe the installation of the impact limiters and personnel barrier. These steps may be omitted as needed.

7.3.2 Preparation for Empty Package Shipment

1. Install the closure plate as follows:

Note:
For empty shipments of the HI-STAR 100 overpack, used metallic seals may be reused.

   a. Remove the Seal Surface Protector from the HI-STAR 100 overpack if necessary.

   b. Perform a contamination survey of the top accessible 12-inches of the HI-STAR 100 Overpack inside surface in accordance with 49CFR173.421(a)(3) [7.1.3]. Verify that the non-fixed contamination levels do not exceed 220,000 dpm/100 cm^2 from beta and gamma radiation and 22,000 pm/100 cm^2 from alpha.

   c. Verify that the HI-STAR 100 Overpack is empty and contains less than 15 gm U-235 in accordance with 49CFR173.421(a)(5) [7.3.1].

   d. Raise and install the closure plate on the HI-STAR 100 overpack. See Figure 7.1.12 for rigging.

   e. Install and torque the closure plate bolts. See Table 7.1.3 for torque requirements.

   f. If necessary, install the vent and drain coverplates.

2. Position the HI-STAR 100 overpack on the transport frame as follows:

   a. Install the HI-STAR 100 overpack buttress plate on the HI-STAR 100 overpack. See Figure 7.1.11 and 7.1.12 for rigging. See Table 7.1.3 for torque requirements.

   b. Downend the HI-STAR 100 overpack in the transport frame. See Section 7.1.2.

   c. Install the removable shear ring segments. See Table 7.1.3 for torque requirements.

3. Perform a final inspection of the HI-STAR 100 overpack as follows:

> **Note:**
> Prior to shipment of the HI-STAR 100 package, the accessible external surfaces of the HI-STAR 100 packaging (HI-STAR 100 overpack, impact limiters, personnel barrier, tie-down, transport frame and transport vehicle) shall be surveyed for removable radiological contamination in accordance with 49CFR173.443(a) [7.1.3] and show less than 2200 dpm/100 cm^2 from beta and gamma emitting sources, and 220 dpm/100 cm^2 from alpha emitting sources.

   a. Perform a final survey for removable contamination. Record the results on the shipping documentation.

   b. Perform a radiation survey of the HI-STAR 100 Overpack and confirm that the radiation levels on any external surface of the overpack do not exceed 0.5 mrem/hour in accordance with 49CFR173.421(a)(2) [7.1.3].

   c. Perform a visual inspection of the HI-STAR 100 overpack to verify that there are no outward visual indications of impaired physical condition and that the package is securely closed in accordance with 49CFR173.428(b) [7.1.3]. Identify any significant indications to the cognizant individual for evaluation and resolution and record on the shipping documentation.

   d. Verify that the HI-STAR 100 overpack neutron shield rupture discs are installed, intact and not covered by tape or other covering.

4. Install the tie-down over the HI-STAR 100 overpack and secure the tie-down bolts. See Table 7.1.3 for torque requirements.

5. *If necessary,* Install the impact limiters as follows:

   a. Install the alignment pins in the bottom of the HI-STAR 100 overpack. See Figure 7.1.11. See Table 7.1.3 for torque requirements.

   b. Using the impact limiter handling frame, raise and position the impact limiter over the end of HI-STAR 100. See Figure 7.1.10.

   c. Install the impact limiter bolts. See Table 7.1.3 for torque requirements.

   d. Repeat for the other impact limiter.

> **Note:**
> The impact limiters cover all the HI-STAR 100 penetrations. The security seals are used to provide tamper detection.

   e. Install a security seal (one each) through the threaded hole in the top and bottom impact limiter bolts. Record the security seal number on the shipping documentation.

f. Perform final radiation surveys of the package surfaces per 10CFR71.47 [7.0.1] and SAR Section 8.1.5.2~~.1~~ and 49CFR173.428(a)(2) [7.1.3].

6. Install the personnel barrier as follows:

a. Rig the personnel barrier as shown in Figure 7.1.9 and position the personnel barrier over the frame.

b. Remove the personnel barrier rigging and install the personnel barrier locks.

c. Transfer the personnel barrier keys to the carrier.

7. Perform a final check to ensure that the package is ready for release as follows:

a. Verify that the receiver has been notified of the impending shipment.

b. Verify that any labels previously applied in conformance with Subpart E of 49CFR172 [7.1.3] have been removed, obliterated, or covered and the "Empty" label prescribed in 49CFR172.450 [7.1.3] is affixed to the packaging in accordance with 49CFR173.428(d) [7.1.3].

c. Verify that the package for shipment is prepared in accordance with 49CFR173.422 [7.1.3]

d. Verify that all required information is recorded on the shipping documentation.

8. Release the HI-STAR 100 System for transport.

7.4 PROCEDURE FOR PREPARING THE HI-STAR 100 OVERPACK FOR TRANSPORT FOLLOWING A PERIOD OF STORAGE

7.4.1 Overview of the HI-STAR 100 System Preparation for Transport ~~F~~following a ~~P~~period of ~~S~~storage

The operations for preparing the loaded HI-STAR 100 Overpack for transport following a period of storage (in excess of one year from the date of completion of HI-STAR 100 Overpack mechanical seal leakage testing) are identical to the later portion of operations required for normal transport of the package as summarized herein. The cask is positioned and the closure plate test port plug, HI-STAR 100 Overpack vent port cover and drain port cover plates are removed. The MSLD is attached and the mechanical seal leakage test is repeated as described in Section 7.1.6. Following successful completion of the leakage tests the closure plate plug and vent and drain port cover plates are reinstalled with new seals. The package is prepared for transport as described in Section 7.1.6.

For the MPC-68F, The HI-STAR 100 overpack vent port cover plate is removed and a gas sample is drawn from the HI-STAR 100 overpack annulus to determine the condition of the MPC confinement boundary. The annulus is vented (as described in Section 7.2.3), evacuated and backfilled with nitrogen gas several times to clear residual helium from the annulus space. The MPC-68F is then leakage tested as described in Section 7.1.6. Following the leakage test of the MPC-68F, the HI-STAR 100 Overpack is prepared for transport as described in Section 7.1.6.

~~7.0.2~~*7.4.2* Preparation for Transport Following a ~~P~~period of Storage

1. Position the HI-STAR 100 Overpack from leakage testing.

Note:
Leakage testing for transport is only required for packages whose metallic seals have not been leakage tested within the last 12 months (in excess of one year from the date of completion of HI-STAR 100 Overpack mechanical seal leakage testing). Step 2 is only required for the MPC-68F. Skip this step if not applicable.

2. If necessary, perform a leakage test of the MPC-68F as follows:

   a. Sample the annulus gas as follows:

      1. Remove the overpack vent port cover plate and attach the backfill tool with a sample bottle attached. See Figure 7.2.3. Store the cover plate in a site-approved location.

      2. Using a vacuum pump, evacuate the sample bottle and backfill tool.

      3. Slowly open the vent port plug and gather a gas sample from the annulus. Reinstall the HI-STAR 100 overpack vent port plug.

b. Evaluate the gas sample and determine the condition of the MPC confinement boundary.

c. If the confinement boundary is intact (i.e., no radioactive gas is measured) then vent the overpack annulus by removing the overpack vent port seal plug (using the backfill tool). Otherwise vent the annulus gas in accordance with instructions from Radiation Protection.

d. Flush the annulus and perform leakage testing of the MPC-68F as follows:

   1. Install the overpack test cover to the overpack vent port as shown on Figure 7.1.27. See Table 7.1.3 for torque requirements.

   2. Evacuate the annulus to below 5 torr and break the vacuum allowing air to fill the annulus space. Repeat this process several times to remove residual helium from the annulus space.

   3. Evacuate the annulus per the MSLD manufacturer's instructions and isolate the vacuum pump from the overpack test cover.

   4. Perform a leakage rate test of MPC-68F per the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be ≤ 5.0E-6 std cc/sec (He) with a minimum test sensitivity less than 2.5E-6 std cc/sec (He).

   5. Disconnect the overpack test cover.

   6. Remove the closure plate test port plug. Discard any used metallic seals.

e. Dry and backfill the overpack annulus as follows:

   1. Load the backfill tool with the HI-STAR 100 overpack vent port plug and the vent port with a new plug seal. Attach the backfill tool to the HI-STAR 100 overpack vent port with the plug removed. See Figure 7.1.28. See Table 7.1.3 for torque requirements.

   2. Attach the Vacuum Drying System to the backfill tool and reduce the HI-STAR 100 overpack pressure to below 3 torr.

Note:
The annulus pressure may rise due to the presence of water in the HI-STAR 100 overpack. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the Vacuum Drying System, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

3. Perform a HI-STAR 100 overpack Annulus Dryness Verification. The overpack annulus shall hold stable vacuum drying pressure of ≤ 3 torr for ≥ 30 minutes.

4. Attach the helium supply to the backfill tool.

5. Verify the correct pressure on the helium supply (pressure set to10+4/-0 psig) and open the helium supply valve.

6. Backfill the HI-STAR 100 overpack annulus to ≥ 10 psig and ≤ 14 psig.

7. Install the overpack vent port plug and torque. See Table 7.1.3 for torque requirements.

8. Disconnect the overpack backfill tool from the vent port.

9. Flush the overpack vent port recess with compressed air to remove any standing helium gas.

3. Leak test the HI-STAR 100 Overpack vent and drain port plug mechanical seals as follows:

a. Install the overpack test cover to the overpack vent port as shown on Figure 7.1.27. See Table 7.1.3 for torque requirements.

b. Evacuate the test cavity per the MSLD manufacturer's instructions and isolate the vacuum pump from the overpack test cover.

c. Perform a leakage rate test of overpack vent port plug per the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be $\leq$ 4.3E-6 std cc/sec (He) with a minimum test sensitivity less than 2.15E-6 std cc/sec (He).

d. Remove the overpack test cover and install a new metallic seal on the overpack vent port cover plate. Discard any used metallic seals.

e. Install the vent port cover plate and torque the bolts. See Table 7.1.3 for torque requirements.

f. Repeat Steps a through e for the overpack drain port.

4. Leak test the overpack closure plate inner mechanical seal as follows:

a. Attach the closure plate test tool to the closure plate test port with the and MSLD attached. See Figure 7.1.29. See Table 7.1.3 for torque requirements.

b. Evacuate the closure plate test port tool and closure plate inter-seal area per the MSLD manufacturer's instructions.

c. Perform a leakage rate test of overpack closure plate inner mechanical seal in accordance with the MSLD manufacturer's instructions. The overpack Helium Leak Rate shall be ≤ 4.3E-6 std cc/sec (He) with a minimum test sensitivity less than 2.15E-6 std cc/sec (He).

d. Remove the closure plate test tool from the test port and install the test port plug with a new mechanical seal. See Table 7.1.3 for torque requirements. Discard any used metallic seals.

5. Verify that the HI-STAR 100 overpack dose rates are within the acceptance requirements listed above.

6. Continue cask loading and preparation for transport as described in Section 7.1.7.

7. The volumetric or multi-layer PT examination of the LTS weld, in conjunction with other examinations which will be performed on this weld (PT of root and final pass, hydrostatic test, and helium leakage test); the use of the ASME Code Section III acceptance criteria; and the additional weld material added to account for potential defects in the root pass of the weld, 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 Chapter 1). 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 ~~in accordance with ANSI N14.6.~~ 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 Hydrostatic Testing

8.1.2.2.1 HI-STAR 100 Containment Boundary

The containment boundary of the HI-STAR Package shall be hydrostatically 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 overpack drain port is used for filling the cavity with water and the vent port for venting the cavity. The written and approved test procedure shall clearly define the test equipment arrangement.

The overpack hydrostatic test *may* ~~shall~~ be performed *at any time during fabrication after the containment boundary is complete.* ~~after the inner shell, bottom plate, and top flange have been welded together, but before the first intermediate shell is attached~~. *Preferably, the hydrostatic 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 the value specified in Chapter 7.

The calibrated test pressure gage installed on the overpack shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure *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, ~~all weld joints~~ *the overpack* shall be ~~visually examined~~ *observed* for leakage. ~~The acceptance criterion shall be zero visible leakage~~ *In particular, the closure plate-to-top forging joint (the only credible leakage point) shall be examined.* If a leak is discovered, the ~~cavity~~ *overpack* shall be emptied and an ~~examination~~ *evaluation* shall be performed to determine the cause of the leakage. Repairs and retest shall be performed until the hydrostatic test acceptance criterion is met.

> Note: If failure of the hydrostatic retest occurs after initial repairs are completed, a nonconformance report shall be issued and root cause and corrective action shall be addressed before further repairs and retest are performed.

After completion of the hydrostatic 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. ~~Liquid penetrant examination of welds shall be performed in accordance with ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NB, Article NB-5350. Any unacceptable areas shall be repaired in accordance with the ASME Code Section III, Subsection NB, NB-4450, and re-examined per the applicable ASME Code as specified in Table 8.1.3.~~ The overpack shall ~~also~~ be required to be hydrotested 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

Hydrostatic testing 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, when field welding of the MPC lid-to-shell weld is completed. The hydrostatic pressure for the test shall be 125 +5,-0 psig, which is 125% of the design pressure of 100 psig. 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 10-minute hold period at the hydrostatic test pressure, and while maintaining a minimum test pressure of 125 psig, the surface of the MPC lid-to-shell weld shall be visually examined for leakage and then 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, the MPC cavity vented (to the pool or the licensee's off-gas system), 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 hydrostatic test shall be repeated until all visual and dye penetrant 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.

Noncontainment 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 noncontainment 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 ~~Bubble~~ Testing of the Neutron Shield Enclosure Vessel

A pneumatic ~~bubble~~ *pressure* test of the neutron shield enclosure vessel shall be performed ~~in accordance with Section V, Article 10, Appendix I, of the ASME Boiler and Pressure Vessel Code~~ 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 rupture disc relief set pressure. The test shall be performed in accordance with approved written procedures.

During the test, the two rupture discs on the neutron shield enclosure vessel shall be removed. One of the rupture disc threaded connections is used for connection of the air pressure line and the other rupture disc connection will be used for connection of the pressure gauge.

Following the introduction of pressurized air into the neutron shield enclosure vessel, a 15 minute ~~minimum soak~~ *pressure hold* time is required. *If the neutron shield enclosure vessel fails to hold pressure,* ~~Following completion of the soak time~~, an approved soap bubble solution shall be applied to *determine the location of the leak.* ~~all enclosure shell return and~~

~~enclosure panel welds. The acceptance criterion for the bubble test shall be no air leakage from any tested weld, as indicated by continuous bubbling of the solution. If air leakage is indicated, the~~ *The leak* ~~weld~~ shall be repaired using weld repair procedures prepared in accordance with the ASME Code Section III, Subsection NF, Article NF-4450. The pneumatic *pressure* ~~bubble~~ test shall be re-performed until no *pressure loss* ~~air leakage~~ 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

~~Upon completion of welding of the inner shell to the bottom plate and top flange, a~~ *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} std cm^3/s (helium). Containment boundary welds shall have indicated leakage rates not exceeding 4.3×10^{-6} std 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.

> Note: If failure of the leakage rate retest occurs after initial repairs are completed, a nonconformance report shall be issued and root cause and corrective action shall be addressed before further repairs and retest are performed.

At the completion of overpack fabrication, helium leakage through the 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} std cm^3/sec (helium) at a minimum test sensitivity of 2.15×10^{-6} std 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 $4.3x10^{-6}$ std cm^3/s (helium) at a minimum test sensitivity of $2.15x10^{-6}$ std cm^3/s (helium).

The leakage testing is performed by evacuating and backfilling the overpack with helium gas to *an appropriate* pressure.~~ of 10 psig +4,-0 psig~~. A helium Mass Spectrometer Leak Detector (MSLD) with a minimum calibrated sensitivity of $2.15x10^{-6}$ std cm^3/s (helium) shall be used in parallel with a vacuum pump and a test cover (see Chapter 7 for details) ~~specifically~~ designed for testing the penetration seals. Starting with the vent *or drain* port plug, the test cover is connected. The cavity on the external side of the ~~vent~~ port plug is evacuated and the vacuum pump is valved out. The MSLD detector measures the leakage rate of helium into the test cavity.~~ If the measured leakage does not exceed a leakage rate of 4.3x10⁻⁶ std cm³/s (helium), the vent port plug seal is acceptable~~. If the leakage rate exceeds a leakage rate of $4.3x10^{-6}$ std 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 $4.6x10^{-6}$ std cm^3/sec, the leakage tests are successful. If the total leakage rate exceeds $4.6x10^{-6}$ std 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

Upon the completion of welding the MPC shell to the baseplate, a confinement boundary weld leakage test shall be performed using a helium MSLD having a minimum calibrated sensitivity of $2.5x10^{-6}$ std cm^3/s (helium) as described in Chapter 9 of the HI-STAR TSAR [8.1.4]. The pressure boundary welds of the MPC canisters shall have indicated leakage rates not exceeding $5x10^{-6}$ std cm^3/s (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.

> Note: If failure of the leakage rate retest occurs after initial repairs are completed, a nonconformance report shall be issued and root cause and corrective action shall be addressed before further repairs and retest are performed.

Leakage testing of the field welded MPC lid-to-shell weld shall be performed following completion of the MPC hydrostatic 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 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-68F. The test is performed with the MPC-68F loaded into the HI-STAR overpack. The HI-STAR overpack annulus is sampled to inspect for radioactive material and then evacuated to a*n appropriate* vacuum ~~of <1 torr~~*condition*. The HI-STAR overpack annulus is then isolated from the vacuum pump. Following ~~a 3-minute~~ *an appropriate* isolation period, the HI-STAR overpack annulus atmosphere is sampled for helium leakage from the MPC-68F. The helium MSLD and test procedure shall have a minimum sensitivity of 2.5×10^{-6} std cm^3/s (helium). The test is considered acceptable if the detected leakage from the MPC does not exceed 5×10^{-6} std cm^3/s (helium). If the acceptance criteria are not met, transport of the MPC-68F 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, Rupture Discs, 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) 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 rupture discs installed in the upper ledge surface of the neutron shield enclosure vessel of the HI-STAR overpack. These rupture discs are provided for venting purposes under hypothetical fire accident conditions in which vapor formation from neutron shielding material degradation may occur. The rupture discs are designed to relieve at 30 psig (± 5 psig). Each manufactured lot of rupture discs shall be sample tested ~~in accordance with a written and approved procedure~~ to verify their point of rupture. The sample test program shall be documented and the test results shall become part of the quality record documentation package.

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 Design Drawings and Bills-of-Material in Chapter 1 specify the crush strength of the aluminum honeycomb materials 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. The honeycomb material crush strength for each block (nominal ± 7%) shall be as specified on the Design Drawings in Section 1.4.

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 ~~(mixed batch)~~ 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.~~and the Bills-of-Material~~. *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 lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

Steel:

The steel plates utilized in the construction of the HI-STAR 100 Package shall be dimensionally inspected to assure compliance with the Design 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 Design Drawing requirements of Section 1.4. Measurements shall be performed in accordance with written and approved procedures. The measurement locations and measurement results shall be documented. 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 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 at the loading facility site to verify the effectiveness of the gamma and neutron shields. This test shall be performed after the HI-STAR 100 Package has been loaded with fuel, drained, sealed, and backfilled with helium.

The neutron and gamma shielding effectiveness tests shall be performed using written and approved procedures. Calibrated neutron and gamma dose meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STAR overpack. Measurements shall be taken at three cross sectional planes and at four points along each plane's circumference. Additionally, four measurements shall be taken at the top of the overpack closure plate. Dose rate measurements shall be documented and become part of the quality documentation package. The average results from each sectional plane shall be compared to the design basis limits for surface dose rates established in Chapter 5. The test is considered acceptable if the actual dose readings are less than the predicted dose rates for the location measured. If dose rates are higher than the predicted design basis dose rates, the HI-STAR 100 Package shall not be placed into transport service until the discrepancy is adequately resolved. See Chapter 7 for details on test performance and dose rate measurement locations.

8.1.5.3 Neutron Absorber Tests

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. Any panel in which ^{10}B loading is less than the minimum allowed per the Design Drawings and Bills-of-Material shall be rejected.

Tests shall be performed using written and approved procedures. Results shall be documented and become part of the HI-STAR 100 Package 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 Design Drawings in Chapter 1. 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

~~Each~~ *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

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, as defined on design drawings, per NB-5300, *NF-5300*, and NG-5300, as applicable. b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements. c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements. d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. 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 and in the Design Drawings. b) ASME Code NB-6000 hydrostatic 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 prior to the transport of the HI-STAR 100 Package containing fuel debris. Acceptance criteria are defined in Subsection 8.1.3.2.

Table 8.1.2 (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 ~~per ANSI N14.6~~. d) A hydrostatic test of the containment boundary in accordance with ASME Code Section III, Subsection NB-6000 and 10CFR71.85(b) shall be performed. ~~during fabrication~~. e) A pneumatic pressure test of the neutron shield enclosure shall be performed during fabrication.	a) None.	a) The rupture discs 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 ~~prior to installation of the intermediate shells~~. 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 quality category. 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 shall be performed after the first fuel loading and reperformed 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 ~~of the HI-STAR overpack.~~	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 Subsection 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) Fit-up test of the HI-STAR overpack rotation trunnions with the transport frame shall be performed. 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 (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 hydrostatic 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 (~~root and~~ final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (~~root and~~ final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (~~root and~~ final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug~~;~~ *and* lift lug baseplate~~, and fuel spacers~~	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-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 Design 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*

U.S. Nuclear Regulatory Commission
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Attachment 6

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Alan I. Soler, being duly sworn, depose and state as follows:

(1) I am Executive Vice President of Holtec International and 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 are is Holtec International Document ID No. HSP-107, *Manufacturing and Testing Procedure for Holtite Neutron Shielding Material*, Revision 3. This document is considered proprietary to Holtec International.

(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), 2.790(a)(4), and 2.790(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).

(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;

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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, 4.b, 4.d, and 4.e, 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 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.

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(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. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(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

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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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STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Dr. Alan I. Soler, 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 22nd day of November, 1999.

Alan Soler

Dr. Alan I. Soler
Holtec International

Subscribed and sworn before me this 22nd day of November, 1999.

Maria C. Pepe

MARIA C. PEPE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2000